

QUESTION # 001:

Given the following:

- A reactor trip has occurred from 100% Rated Thermal Power.
- E-0, "Reactor Trip or Safety Injection", is in progress at Step 2: Verify Turbine Trip.
- HP turbine impulse pressure is 105 psig and lowering.
- Turbine Stop Valve SV-1 CLOSED light is LIT.
- Turbine Stop Valve SV-2 CLOSED light is LIT.
- Turbine Stop Valve Closed Bistable light 44907-1107, Turbine Left Stop VLV Closed, is LIT.
- Turbine Stop Valve Closed Bistable light 44907-1108, Turbine Right Stop VLV Closed, is NOT LIT.

In accordance with E-0, which of the following is the NEXT REQUIRED action?

- a. Stop both EH oil pumps.
- b. Manually trip the Turbine.
- c. Check Bus 5 and Bus 6 energized.
- d. Initiate a Main Steamline Isolation.

Answer

c.

QUESTION # 002

Given the following:

- The unit is in MODE 3.
- PRZR Pressure Control Channel Selector switch is in the 2-3 position.
- PRZR Level Control Channel Selector switch is in the 2-3 position.
- Pressurizer pressure instruments:
 - PI-431 and PI-449 read off-scale low
 - PI-429 and PI-430 read 2185 psig and are lowering slowly
- Pressurizer level instruments:
 - LI-428, Channel 3, is reading 75% and rapidly rising
 - LI-426 and LI-427 are reading 35% and are stable
- Charging Pump 'A' is in automatic and speed is at minimum.
- Annunciator 47043-J, Charging Pump In Auto High/Low Speed, is LIT.

What event is occurring?

- a. PR-2A, PRZR PORV, has failed OPEN.
- b. Pressurizer Pressure Master Controller failed High.
- c. A LEAK has developed in the bellows at the upper tap for LT-428.
- d. A Safety Injection has occurred due to the failure of PI-431 and PI-449 LOW.

Answer

c.

QUESTION # 003:

Given the following:

- ES-1.2, "Post LOCA Cooldown and Depressurization," is in progress following a small break LOCA.
- SI pumps are injecting into the core.
- RCS wide range temperature is 540°F.
- RCS subcooling is 56°F.
- Pressurizer heater control switches are in the OFF position.
- Adverse Containment conditions are NOT present.
- Pressurizer level is 0%.
- RXCP B is running.
- The Operator is directed to throttle open PS-1B, PRZR Spray Valve to refill the pressurizer.

How can the operator control PS-1B AND what is the reason for refilling the pressurizer?

	<u>Control</u>	<u>Reason</u>
a.	Manually Close PS-1A and then adjust the Pressurizer Spray Master Controller Automatic Setpoint	Establish Pressurizer level to allow automatic RCS pressure control
b.	Manually Close PS-1A and then adjust the Pressurizer Spray Master Controller in Manual	Raise RCS subcooling by causing the Safety Injection Accumulators to inject
c.	With PS-1B in Automatic adjust the Automatic Setpoint on the PS-1B controller	Prevent rapid RCS pressurization when Safety Injection flow exceeds break flow
d.	With PS-1B in Manual adjust the valve position of PS-1B Safety Injection flow	Add to the RCS inventory by reducing break flow and raising

Answer

d.

QUESTION # 004

Given the following:

- The plant has experienced a Large Break LOCA.
- The crew has entered ECA-1.3, "Containment Sump Blockage"
- All attempts to establish containment sump recirculation have failed.
- Plant parameters and equipment status:

<u>Parameter</u>	<u>Value</u>	<u>Trend</u>
RWST Level	<3%	Stable
Boric Acid Storage Tanks	<9%	Stable
Core Exit Thermocouples	900°F	Stable
RCS Wide Range Pressure	30 psig	Stable
Containment Pressure	5 psig	Slowly Lowering
Hottest Wide Range RCS Hot Leg	300°F	Stable

<u>Equipment</u>	<u>Status</u>
SI Pumps	PULLOUT
RHR Pumps	PULLOUT
ICS	PULLOUT
RXCPs	PULLOUT
Charging Pumps 'A' & 'B'	Running at Max (Suction from VCT)

- The crew has begun to DEPRESSURIZE All Intact Steam Generators to Atmospheric Pressure.
- The crew is waiting for concurrence from the Emergency Director to place the RHR system in service.
- The crew is maintaining RCS Heat Removal by continuing to dump steam from the intact SGs.

What is the basis for dumping steam from the intact SGs?

- Maintain a differential temperature between RCS Hot and Cold legs to promote natural circulation.
- Maintain proper thermal stratification in the SGs in the event of a SG Tube Rupture.
- To ensure heat removal via Reflux Boiling/Natural Circulation.
- To improve net positive suction to the RHR pumps.

Answer

c.

QUESTION # 005

Given the following:

- The unit has been at 100% Rated Thermal Power for the last 42 days.
- Annunciator 47013-I, RXCP A Seal Leakoff Flow High/Low, is LIT.
- Annunciator 47015-I, RXCP A Standpipe High/Low, is LIT.
- Reactor Coolant Pump 'A' #1 seal leakoff flow indicates 0.9 GPM.
- Operators determine that Standpipe Level is HIGH.
- Reactor Coolant Drain Tank (RCDT) level is rising faster than normal.

Which of the following has occurred?

- a. #1 Seal has failed open
- b. #2 Seal has failed open
- c. #3 Seal blockage
- d. #1 Seal blockage

ANSWER

b.

QUESTION # 006

Given the following:

- The unit is at 100% Rated Thermal Power.
- VCT level transmitter LT-112 (24015) fails high (100%).
- Annunciator 47043-L, VCT Level High/Low, is LIT.
- The operator is directed to perform manual makeups to the VCT as required to maintain VCT Level

Why does the operator take manual action to maintain VCT level?

To prevent . . .

- a. the charging pumps from losing suction and cavitating because auto makeup will NOT maintain VCT level.
- b. continuous automatic makeup caused by LD-27, VCT/Holdup Tank Divert Valve, shifting to DIVERT.
- c. LD-14, LD Demin High Temp Divert Valve, from shifting to DIVERT caused by a rise in letdown flow.
- d. losing control of plant reactivity caused by the automatic shifting of charging pump suction to the RWST as VCT level lowers.

ANSWER

a.

QUESTION # 007

Given the following:

- A heat-up is in progress following a refueling shutdown.
- Highest cold leg temperature indicates 205°F.
- RXCP 'A' is running.
- Steam Generator 'A' & 'B' narrow range levels are greater than 20%.
- RHR is being aligned for Split Train Mode per NOP-RHR-001, "Residual heat Removal System Operation"
- SI-350B, CNTMT Sump B To RHR Pump B, failed to OPEN when directed to cycle the valve per NOP-RHR-001.

What additional failure would require action to place the plant in MODE 5 if NOT corrected within ONE hour?

- a. SI-4A, RWST Supply to SI Pumps, failed CLOSED
- b. SI-300A, RWST Supply to RHR pump A, failed CLOSED
- c. RHR-1A, RCS Loop A Supply to RHR Pumps, failed CLOSED
- d. ICS-2B, CNTMT Spray Pump B Suction From The RWST, failed CLOSED

ANSWER

b.

QUESTION # 008

Given the following:

- The unit is operating at 100% Rated Thermal Power.
- A malfunction in the Component Cooling Water System has occurred.
- PPCS Component Cooling Temperature T0621A is 105°F and rising.

From the list below complete the following. If the component cooling water heat exchanger outlet temperature rises to (1) , the correct operator actions would be to (2).

(1)

(2)

- | | |
|----------|--|
| a. 110°F | manually start the standby component cooling water pump |
| b. 120°F | trip the reactor and trip both Reactor Coolant pumps |
| c. 135°F | isolate CC-300, Letdown Heat Exchanger Inlet Isolation Valve |
| d. 145°F | isolate CC-610A/B, RXCP Therm Barr Comp Cooling Return Valve |

ANSWER

b.

QUESTION # 009

Given the following:

- The unit is at 100% Rated Thermal Power.
- Pressurizer Pressure Control is in Automatic controlling pressure at 2235 psig.
- PRZR Pressure Channel Control Selector switch is in the 2-1 position.
- PT-429, Red Channel Pressurizer Pressure has Failed LOW.

With NO Operator action what prevents RCS pressure from exceeding the Technical Specification Pressure Safety Limit?

- a. The Pressurizer Safeties will OPEN
- b. ONLY PR-2A, Pressurizer PORV, will OPEN
- c. ONLY PR-2B, Pressurizer PORV, will OPEN
- d. BOTH PR-2A and PR-2B, Pressurizer PORVs will OPEN

ANSWER

a.

QUESTION # 010

Given the following:

- The unit was operating at 100% Rated Thermal Power when a Small Break LOCA occurred.
- Safety Injection initiated but the reactor trip breakers FAILED TO OPEN.
- The crew transitioned to FR-S.1, "Response to Nuclear Power Generation/ATWS".
- Subsequent actions by the Operators have been **UNSUCCESSFUL** in lowering Reactor NI power less than 20% before an Operator identifies that RCS subcooling is 15°F and degrading rapidly.

What action is directed in FR-S.1 regarding the operation of the Reactor Coolant Pumps (RXCPs) AND what is the reason for the action?

<u>Action</u>	<u>Reason</u>
a. Maintain the RXCPs operating	Provide forced flow through the core for heat removal as the core voids.
b. Maintain the RXCPs operating	Reduce the effect of core voiding on the Source Range Nuclear Instrument indication.
c. Stop the RXCPs	Prevent excessive depletion of RCS inventory to maximize time the RWST is available.
d. Stop the RXCPs	Prevent damage to the pumps which allows for their use later in heat removal from the core.

ANSWER

a.

QUESTION # 011

Given the following:

- Reactor Power was originally at 100% Rated Thermal Power.
- Pressurizer level lowered rapidly and is currently 2% and lowering.
- A reactor trip and loss of Off-Site power occurred.

Which diagnosis indicates a Steam Generator (SG) tube rupture has occurred?

- a. Reactor Thermal Power was rising before the reactor trip.
- b. AFW flow to the ruptured SG is greater than AFW flow to intact SG.
- c. SG pressure of the ruptured SG is rising in an uncontrolled manner.
- d. Indicated steam flow was greater than feed flow before the reactor trip.

ANSWER

d.

QUESTION # 012

Given the following:

- The crew is performing FR-H.1, "Response to Loss of Secondary Heat Sink"
- The crew has OPENED both Pressurizer PORVs
- The crew has verified MCC-5262 energized by checking C-13, Condensate Bypass LP FW Heater, or Air Compressor 1A energized

Why is MCC-5262 verified energized?

- a. Check a supply of compressed air is available for control of PR-2A and PR-2B, Pressurizer PORVs.
- b. Check power is available to locally operate FW-2A and FW-2B, Feed Water Pump Discharge Isolation Valves.
- c. Check that the Turbine Seal Oil system is operating to prevent the Hydrogen concentration in the Turbine from reaching explosive levels.
- d. Check that a Waste Gas compressor is available to maintain Pressurizer Relief Tank Conditions to prevent rupturing of the Pressurizer Relief Tank Rupture Disk.

ANSWER

a.

QUESTION # 013

Given the following:

- A loss of Off-Site power has occurred.
- Bus 6 is LOCKED OUT.
- Diesel Generator 'A' has tripped.
- Annunciator 47091-B, Diesel Gen A Mech Lockout, is LIT
- The local operator reports each of the alarms listed below are LIT and Diesel Generator 'A' Primary Tank Air Pressure is 190 psig.

Which of the following alarms, if it alone were cleared, would allow a restart of Diesel Generator 'A'?

- a. Diesel Generator annunciator DR101-11, High Water Temp.
- b. Diesel Generator annunciator DR101-21, Low Lube Oil Level.
- c. Diesel Engine Control Panel D-1A L7, Jacket Water Pressure, lamp.
- d. Diesel Engine Control Panel D-1A L10, Low Air Pressure System #2 (Primary), lamp.

ANSWER

c.

QUESTION # 014

Given the following:

The unit was operating at 100% Rated Thermal Power when the following indications were observed:

- Service Water (SW) Header Train 'A' indicated 78 psig and lowering
- Service Water Pumps 1A1, 1A2, and 1B1 are running
- Circulating Water Pump 'A' TRIPPED
- Circulating Water Pump 'B' TRIPPED
- SW-4A, SW Header A to Turbine Bldg Hdr, is OPEN
- SW-4B, SW Header B to Turbine Bldg Hdr, is CLOSED

Which of the following could cause ALL the above indications?

- a. SW Strainer differential pressure indicates 6 psid.
- b. A SW leak has occurred in the turbine building.
- c. A Reserve Auxiliary Transformer Lockout
- d. Forebay water level indicates 46%.

ANSWER

b.

QUESTION # 015

Given the following:

- The unit is at 100% Rated Thermal Power.
- IA-101, Instrument Air to Containment Isolation valve, is inadvertently CLOSED and NOT RE-OPENED.
- A small air leak exists in the Containment air header.

If air pressure is NOT restored to containment, which of the following will cause a reactor trip?
(Assume NO operator action)

- a. Letdown isolation will result in HIGH Pressurizer level.
- b. Closure of Main Feed Regulating Valves will result in LOW SG level.
- c. Opening of Pressurizer Power Operated Relief Valves will result in LOW Pressurizer pressure.
- d. Containment Fan Coil Units Emergency Discharge Dampers opening will result in a Reactor Coolant Pump TRIP due to overheating.

ANSWER

a.

QUESTION # 016

Given the following:

- The unit was operating at 45% Rated Thermal Power when the reactor automatically tripped.
- Safety Injection automatically initiated.
- The Crew diagnosed a LOCA outside of Containment and transitioned to ECA-1.2, "LOCA Outside Of Containment".
- The LOCA was isolated when the crew CLOSED SI-302A, RHR Pump A Injection To Reactor Vessel.
- RWST level continued to lower until the crew CLOSED SI-300A, RWST Supply To RHR Pump A.
- Plant Parameters after SI-300A, RWST Supply to RHR Pump A, was closed were:

<u>Parameter</u>	<u>Value</u>	<u>Trend</u>
Pressurizer Pressure	2210 psig	Stable
Pressurizer Level	30%	Rising
Cold Leg Temperatures	450°F	Rising
RWST Level	273,000 gal (Usable)	Stable
Total AFW Flow	240 gpm	Slowly Lowering
Steam Generator 'A' Pressure	410 psig	Rising
Steam Generator 'B' Pressure	400 psig	Rising
Steam Generator 'A' Narrow Range Level	11%	Rising
Steam Generator 'B' Narrow Range Level	15%	Rising

The following valves have been closed and tagged closed to isolate the leak

- RHR-6A, RHR Pump Discharge Valve
- RHR-7A, Heat Exchanger Isolation
- RHR-100A, Heat Exchange Bypass Line

Which of the following is NOT able to meet its Technical Specification required surveillances at this time?

- RWST
- RCS Loop 'A'
- ECCS SI Train 'A' subsystem
- ECCS RHR Train 'B' subsystem

ANSWER

c..

QUESTION # 017

Given the following:

- The unit was at 100% Rated Thermal Power when a large break LOCA occurred.
- The ECCS system is functioning as designed.
- Reactor Coolant Pumps are in PULLOUT
- Current Plant Parameters:

<u>Parameter</u>	<u>Value</u>	<u>Trend</u>
Wide Range RCS Pressure	25 psig	Stable
Core Exit Thermocouples	220°F	Stable
RCS Cold Leg Loop Temperatures	150°F	Stable
Steam Generator Pressures	400 psig	Stable
Steam Generator 'A' Narrow Range Level	2%	Slowly Rising
Steam Generator 'B' Narrow Range Level	4%	Slowly Rising
Total AFW Flow to Both Steam Generators	220 gpm	Stable
Containment Pressure	25 psig	Slowly Lowering
Pressurizer Level	0 %	-----
RVLIS Train 'A' & 'B'	85%	Slowly Lowering
RCS Void Fraction Train 'A' & 'B'	-10%	Stable
RWST Level	50%	Slowly Lowering

What is the current mechanism by which core decay heat is being removed?

- a. AFW flow.
- b. RCS break flow.
- c. forced two phase flow through the core.
- d. heat transfer to Component Cooling from RHR.

ANSWER

b.

18. Given the following:

- The unit is operating at 100% Rated Thermal Power.
- The crew entered AOP-EG-001, "Abnormal Grid Conditions" when 22 OCB Tripped OPEN and grid instability caused the 138KV voltage to lower to 142 KV.

Based on plant Operating Experience what controller operation may occur and what action is directed by AOP-EG-001 to mitigate the consequences of the controller operation?

<u>Controller Operation</u>	<u>Operation</u>
a. Turbine Control Valve Cycling	Disengage the Valve Position Limiter
b. SW-1306A & SW-1306B, CC HX Temp Control Valves, fail OPEN	Position LD-14, LD Demin High Temp Divert Valve, to V.C.TK
c. C-701, Condensate Recirculation Control Valve, fails CLOSED	Take manual control of C-701 in the Control Room
d. Generator #1 Voltage Regulator maintains a constant output voltage causing MVARs to LOWER	Raise Generator #1 Base Adjuster

ANSWER

b.

QUESTION # 019

Given the following:

- The reactor spuriously tripped from 100% Rated Thermal Power.
- All of the control rods in Control Bank 'B' remained fully withdrawn after the reactor trip signal.
- CVC-440, Emergency Boration To Charging Pumps, will NOT OPEN from the Control Room.
- The Unit Supervisor directs the Operator to perform the emergency boration per the RNO for failure of CVC-440 to operate from the control room.

Which of the following component manipulations states the correct system response when establishing the emergency boration flow path?

<u>Component Manipulation</u>	<u>System Response</u>
a. OPEN CVC-403/CV-31092, Boric Acid To Blender	Directs boric acid to the Boric Acid Blender and isolates Reactor Makeup Water from the Boric Acid Blender by causing MU-1022, Reactor Makeup Water to Blender, to CLOSE on interlock
b. CLOSE CVC-405, Boric Acid Emergency Suct Isol	Isolate boric acid flow to the Boric Acid Blender and establishes a boric acid flow path directly from the Boric Acid Storage Tanks to suction of the charging pumps
c. CLOSE CVC-406/CV-31094, BA Blender to VCT	Isolates boric acid flow from the Boric Acid Blender to the VCT and causes CVC-405, Boric Acid to Emergency Suct Isol, to OPEN on interlock
d. OPEN CVC-408/CV-31093, BA Blender To Charging Pumps	Stops a Boric Acid Flow Deviation from automatically CLOSING CVC-408, BA Blender To Charging Pumps

ANSWER

d.

QUESTION # 020

Per AOP-RC-004, "Steam Generator Tube Leak", which of the following is the LARGEST stable RCS to Steam Generator Leak rate which does NOT require isolation of the leaking Steam Generator?

(Stable is defined as a constant rate for greater than 60 minutes)

A stable leak rate of . . .

- a. 10 gallons per day.
- b. 25 gallons per day.
- c. 80 gallons per day.
- d. 150 gallons per day.

ANSWER

c.

QUESTION # 021

Given the following:

- Power is at 7% by Nuclear Instruments during a plant startup.
- Intermediate Range NIS channel N35 Instrument Power has FAILED.
- The following action of AOP-MISC-001, "Response to Instrument Failure", for the failed channel was completed:
 - The Channel Selector switch on the Startup Rate Section of the Comparator and Rate Drawer was POSITIONED to N36.
- The Level Trip switch on the N35 drawer was POSITIONED to BYPASS for corrective maintenance.

Which of the following list of Annunciators and Permissive Lights would be consistent with these plant conditions and switch manipulations?

1. 47033-L, NI System SR/IR Trip Bypassed
2. 47032-N, IR High Flux Rod Stop
3. Permissive Light 44905-0602, Auto Rod WDL Block P-2
4. Permissive Light 44905-0202, IR Blocked

- a. 1 and 3 LIT
- b. 1 and 4 LIT
- c. 2 and 3 LIT
- d. 2 and 4 LIT

ANSWER

a.

QUESTION # 022

Which of the following is an operational implication associated with Area Radiation Monitors R-40 and R-41, Containment Hi Level Radiation Monitors?

R-40 and R-41 contain . . .

- a. a GM tube with a high level of sensitivity to detect RCS leakage as required by Technical Specifications.
- b. an Ion Chamber that is required post-accident and is used to determine Adverse Containment conditions.
- c. a Scintillation detector with high sensitivity which has installed insulation to allow for operation during a Design Basis Accident.
- d. a Solid State Alpha detector which can discriminate in areas with high background radiation and initiates a Containment Vent Isolation.

ANSWER

b.

QUESTION # 023

Given the following:

- The unit is at 100% Rated Thermal Power.
- The Control Room is required to be evacuated due to a fire in the Alternate Fire Zone.
- The crew is performing the actions of AOP-FP-002, "Fire in Alternate Fire Zone."
- The reactor failed to trip from the Control Room.

What actions are required to ENSURE the reactor is tripped per AOP-FP-002?

- a. Locally place Rod Drive MG Set Motor & Generator Circuit Breaker Control Switches to TRIP.
- b. Locally TRIP the reactor at the DSP using the DSP Manual Reactor Trip Pushbutton.
- c. Locally OPEN the reactor trip and bypass breakers.
- d. Locally OPEN bus 33 and 43 supply breakers.

ANSWER

c.

QUESTION # 024

Given the following:

- FR-C.1, "Response To Inadequate Core Cooling" is in progress.
- Both RXCPs were stopped per FR-C.1 in preparation for SG depressurization to atmospheric pressure.
- Actions to reduce Core Exit Thermocouple temperatures have been unsuccessful.
- FR-C.1 directs the subsequent re-start of the RXCPs in available loops.

What is the purpose of the RXCP re-start?

- a. To ensure the SI Accumulators are fully discharged and the coolant is circulated.
- b. To raise RCS pressure to collapse voids which will allow Safety Injection Pump flow.
- c. To force two phase or single phase flow through the core to provide temporary cooling.
- d. To use RXCP discharge pressure to provide PRZR spray flow to reduce RCS pressure.

ANSWER

c.

QUESTION # 025

Given the following:

- The unit is at 100% Rated Thermal power.
- The following annunciators are LIT:
 - TLA-15, RMS Above Normal
 - 47012-B, High Radiation Indication Alert
- Radiation Monitors indicated rising radiation levels in the following sequence:
 - 1st R-9, RCS Letdown Monitor
 - 2nd R-4, Charging Pump Room Monitor
 - 3rd R-13 and R-14, Aux Bldg Vent Exhaust Monitors
- All other Radiation Monitors indicate normal radiation levels

What event does the above sequence of radiation monitor alarms indicate?

- a. LOCA outside containment
- b. High Reactor Coolant Activity
- c. RCS Leak in CVCS letdown line
- d. Charging Pump Relief Valve Lifting

ANSWER

b.

QUESTION # 026

Given the following:

- Small Break LOCA has occurred.
- The crew is performing ES-1.2, "Post LOCA Cooldown and Depressurization".
- RCS pressure is 350 psig and stable.
- Both RHR pumps were stopped and placed in AUTO as directed in ES-1.2.
- The LOCA is still in progress.

After the Safety Injection signal was RESET, the RHR pumps were STOPPED to . . .

- a. conserve RWST inventory.
- b. reduce the rate of RCS cooldown.
- c. prevent damage to the RHR pumps.
- d. minimize component cooling heat load.

ANSWER

c.

QUESTION # 027

Given the following:

- During implementation of FR-P.1, "Response To Imminent Pressurized Thermal Shock Condition", SI-20B, Accumulator 'B' Isolation Valve, failed to CLOSE.
- The Unit Supervisor directs NG-108B, Nitrogen Supply to Accumulator 'B', OPENED AND then NG-110, ACMTR VENT Control, OPENED for venting of Accumulator 'B'.

How does an operator OPEN NG-108B AND NG-110?

NG-108B

- Locally in the Gas Bottle Storage area
- Placing its control switch on Mechanical Vertical 'A' to OPEN
- Adjusting its controller on Mechanical Control Console 'C' to 100%
- Placing its control switch on Mechanical Control Console 'C' to OPEN

NG-110

- Locally in the Gas Bottle Storage area
- Locally in the blowdown tank area
- Placing its control switch on Mechanical Vertical 'A' to OPEN
- Adjusting its controller on Mechanical Control Console 'C' to 100%

ANSWER

d.

QUESTION # 028

What is a function of the Reactor Coolant Pump Flywheel?

- a. Ensures sufficient flow through the core during normal operations to maintain the DNB margin.
- b. Prevent overheating of RXCPs seals in an idle loop due to reverse rotation of the Reactor Coolant Pump.
- c. Maximize loop flow during natural circulation by ensuring sufficient starting inertia to prevent reverse rotation.
- d. Provide inertia for a slow coast down or reduction of reactor coolant flow so that DNB margin is maintained.

ANSWER

d.

QUESTION # 029

Given the following:

- The unit has been operating at 75% Rated Thermal Power for the past 113 hours
- Generator output is 450 MW electric
- Rod Control is in AUTO
- Charging Pump 'A' is in AUTO at 25% demand
- Charging Pump 'B' is in MANUAL at 25% demand
- CC-302, Letdown Heat Exchanger Temperature Control Valve, is in AUTO and set at 120°F
- Tavg is at program value for plant conditions
- Temperature Element TE-401B, Reactor Coolant Loop B-2 Cold Leg Temperature, FAILS HIGH (Red Channel)

With NO operator action, what is a control AND reactor protection function that are affected by the failure of TE-401B?

<u>Control Function</u>	<u>Reactor Protection Function</u>
a. Turbine Runback	Bistable for RCS Loop 'B' Tavg Low Low will be TRIPPED
b. Charging flow will RISE	Loop 'B' Channel 1 Over Temperature Delta Temperature setpoint will LOWER
c. Control Bank 'D' control rods will step OUTWARD	Loop 'B' Channel 1 Over Pressure Delta Temperature setpoint will LOWER
d. CC-302, Letdown Heat Exchanger Temperature Control Valve, will OPEN	Bistable for Red Channel HI Tavg will be TRIPPED

ANSWER

b.

QUESTION # 030

How does the Main Steam Reheat and Heater Drain System design prevent an uncontrolled RCS cooldown which could result in an automatic Safety Injection when the reactor trips?

- a. The MSR reheat inlet valves are downstream of the Turbine Stop Valves, therefore, when the turbine trips steam is isolated to the MSRs.
- b. A turbine trip causes an isolation of steam to the inlet of the MSR by isolating instrument air to MS-201A1/A2/B1/B2, Reheater Steam Control Valves.
- c. A reactor trip signal causes isolation of MSR Drain valves by de-energizing the solenoid valve which isolates instrument air to the MSR Drain valves, these valves then fail CLOSED.
- d. Differential pressure between the Reheater Drain Tank and Feedwater Heater 15A/B greater than 10 psid CLOSES HD-11A1/A2/B1/B2, Reheater Drain Tank to Feedwater Heater 15A/B isolating steam flow from the MSR.

ANSWER

b.

QUESTION # 031

Given the following:

- A plant cooldown to MODE 5 is in progress.
- The RCS is at 335°F and 350 psig.
- Pressurizer bubble exists, with heaters and spray used for pressure control.
- RHR is aligned for split train operations per NOP-RHR-001, "Residual Heat Removal Operation."
- RHR-8A, RHR Heat Exchanger A Flow Control Valve is at 60% open.
- An air leak develops on RHR-8A, RHR Heat Exchanger 'A' Flow Control Valve, completely depressurizing the actuator.
- RHR, RCS, Charging and Letdown flows are stable prior to the failure.
- RHR/CVC Spectacle Flange is OPEN.

With NO operator action, what is an effect of the complete depressurization of RHR-8A, valve actuator?

- a. Regen Hx Letdown Temp (TI-127) LOWERS due to total flow to the Regenerative Heat Exchanger LOWERING.
- b. RCS cooldown rate LOWERS when RHR-8A, RHR Heat Exchanger 'A' Flow Control Valve, CLOSES.
- c. Charging Pump in AUTO speed RISES as level in the Pressurizer LOWERS due to total Letdown flow RISING.
- d. Letdown Hx Outlet Flow (FI-134) LOWERS due to the LOWERING of the differential pressure between RHR and Letdown.

ANSWER

d.

QUESTION # 032

Given the following:

- The plant is in MODE 1.
- A Bus 5 Lockout occurs.
- The plant is immediately manually tripped due to unstable conditions.
- During the performance of the Immediate Actions of E-0, "Reactor Trip or Safety Injection", a spurious Train 'A' Safety Injection signal was received.

Which of the following is the correct display of the Containment Isolation Active Panel in the Control Room following the spurious Train 'A' Safety Injection signal?

(Pictures of the Containment Isolation Active Panel on the next pages)

- a. See Attached Page for selection A
- b. See Attached Page for selection B
- c. See Attached Page for selection C
- d. See Attached Page for selection D

ANSWER

a.

A

CNTMT ISOL ACTIVE

	1	2	3	4	5	6	7
1	LD ORIF 1A LD-4A 0101	LD ORIF 1B LD-4B 0102	LD ORIF 1C LD-4C 0103			DDT VENT WG-311 0106	DDT ISOLATION MD(R)-323A 0107
2		LD HX FLOW LD-6 0202		EXCS LD HX CC-653 0204	VCT VENT CVC-54 0205	DDT VENT WG-310 0206	DDT ISOLATION MD(R)-323B 0207
3	S/G 1A BD BT-2A 0301	S/G 1B BD BT-2B 0302	S/G 1A SAMPLE BT-31A 0303	S/G 1B SAMPLE BT-31B 0304			SEAL WATER LEAKOFF CVC-211 0307
4	S/G 1A BD BT-3A 0401	S/G 1B BD BT-3B 0402	S/G 1A SAMPLE BT-32A 0403	S/G 1B SAMPLE BT-32B 0404		ACMTR N ₂ SPLY NG-107 0406	SEAL WATER LEAKOFF CVC-212 0407
5	PRZR STM SMPL RC-402 0501	PRZR LIQ SMPL RC-412 0502	RCS HOTLEG B SAMPLE VALVE RC-422 0503				
6	PRZR STM SMPL RC-403 0601	PRZR LIQ SMPL RC-413 0602	RCS HOTLEG B SAMPLE VALVE RC-423 0603	PRT GAS ANZR MG(R)-512 0604	PRT GAS ANZR MG(R)-513 0605	PRT N ₂ SPLY NG-302 0606	PRT MAKE-UP MU1010-1 0607
7	H ₂ VENT ISOL LOCA 2B 0701	H ₂ RCMBR TO CNTMT LOCA 201B 0702					
8	H ₂ RCMBR LOCA 100B 0801	H ₂ DILUTE SA7003B 0802			R-11/12 SAMPLE ISOL AS-32 0805	CNTMT SPRAY TEST LINE ICS-201 0806	CNTMT SPRAY TEST LINE ICS-202 0807
9	VACUUM BKR VB-10A 0901	VACUUM BKR VB-10B 0902					
10						CONTAINMENT PURGE RBV-2 1006	CONTAINMENT VENT RBV-3 1007
11	RCDT GAS ANZR MG(R)-503 1101	RCDT VENT MG(R)-509 1102	RCDT PUMPS RC-507 1103	CONTAINMENT SUMP PUMP MD(R)-134 1104	R11/12 SAMPLE ISOL AS-1 1105	CONTAINMENT PURGE RBV-1 1106	CONTAINMENT VENT RBV-4 1107
12	RCDT GAS ANZR MG(R)-504 1201	RCDT VENT MG(R)-510 1202	RCDT PUMPS RC-508 1203	CONTAINMENT SUMP PUMP MD(R)-135 1204	R11/12 SAMPLE ISOL AS-2 1205	CONTAINMENT PURGE TAV-12 1206	CONTAINMENT VENT RBV-5 1207

11

CNTMT ISOL ACTIVE

	1	2	3	4	5	6	7
1	LD ORIF 1A LD-4A 0101	LD ORIF 1B LD-4B 0102	LD ORIF 1C LD-4C 0103			DDT VENT WG-311 0106	DDT ISOLATION MD(R)-323A 0107
2		LD HX FLOW LD-6 0202		EXCS LD HX CC-653 0204	VCT VENT CVC-54 0205	DDT VENT WG-310 0206	DDT ISOLATION MD(R)-323B 0207
3	S/G 1A BD BT-2A 0301	S/G 1B BD BT-2B 0302	S/G 1A SAMPLE BT-31A 0303	S/G 1B SAMPLE BT-31B 0304			SEAL WATER LEAKOFF CVC-211 0307
4	S/G 1A BD BT-3A 0401	S/G 1B BD BT-3B 0402	S/G 1A SAMPLE BT-32A 0403	S/G 1B SAMPLE BT-32B 0404		ACMTR N ₂ SPLY NG-107 0406	SEAL WATER LEAKOFF CVC-212 0407
5	PRZR STM SMPL RC-402 0501	PRZR LIQ SMPL RC-412 0502	RCS HOTLEG B SAMPLE VALVE RC-422 0503				
6	PRZR STM SMPL RC-403 0601	PRZR LIQ SMPL RC-413 0602	RCS HOTLEG B SAMPLE VALVE RC-423 0603	PRT GAS ANZR MG(R)-512 0604	PRT GAS ANZR MG(R)-513 0605	PRT N ₂ SPLY NG-302 0606	PRT MAKE-UP MU1010-1 0607
7	H ₂ VENT ISOL LOCA 2B 0701	H ₂ RCMBR TO CNTMT LOCA 201B 0702					
8	H ₂ RCMBR LOCA 100B 0801	H ₂ DILUTE SA7003B 0802			R-11/12 SAMPLE ISOL AS-32 0805	CNTMT SPRAY TEST LINE ICS-201 0806	CNTMT SPRAY TEST LINE ICS-202 0807
9	VACUUM BKR VB-10A 0901	VACUUM BKR VB-10B 0902					
10						CONTAINMENT PURGE RBV-2 1006	CONTAINMENT VENT RBV-3 1007
11	RCDT GAS ANZR MG(R)-503 1101	RCDT VENT MG(R)-509 1102	RCDT PUMPS RC-507 1103	CONTAINMENT SUMP PUMP MD(R)-134 1104	R11/12 SAMPLE ISOL AS-1 1105	CONTAINMENT PURGE RBV-1 1106	CONTAINMENT VENT RBV-4 1107
12	RCDT GAS ANZR MG(R)-504 1201	RCDT VENT MG(R)-510 1202	RCDT PUMPS RC-508 1203	CONTAINMENT SUMP PUMP MD(R)-135 1204	R11/12 SAMPLE ISOL AS-2 1205	CONTAINMENT PURGE TAV-12 1206	CONTAINMENT VENT RBV-5 1207

11

CNTMT ISOL ACTIVE11

QUESTION #033

Given the following:

- The Plant is in MODE 3 when a Design Basis LOCA occurs
- The crew has transitioned to E-1, "Loss of Reactor or Secondary Coolant"
- RWST level is 55% and slowly lowering
- The STA informs the Unit Supervisor of an ORANGE path for CONTAINMENT due to BOTH Internal Containment Spray pumps tripping on overcurrent
- The crew transitions and COMPLETES FR-Z.1, "Response to High Containment Pressure"
- The ORANGE path for CONTAINMENT still exists
- NO other ORANGE or RED paths exists

Which of the following states the expected response of Containment Pressure at the completion of FR-Z.1?

Containment pressure will . . .

- a. not exceed design pressure as long as four Containment Fan Coil Units are operating to provide heat removal for the steam produced from the Design Basis LOCA.
- b. continue to rise until an equilibrium is established between steam production from the Design Basis LOCA and the rate of heat removal provided by the containment dome fans.
- c. slowly lower because the combination of one train of Containment Fan Coil Units and Shield Building Ventilation are sufficient to lower containment pressure during a Design Basis LOCA.
- d. exceed Containment Design pressure unless one Internal Containment Spray Pump can be started, the heat removal capacity of the containment fan coil units and containment dome fans are not sufficient during a Design Basis LOCA.

ANSWER

a.

QUESTION # 034

Given the following:

- The unit is at 100% Rated Thermal Power.
- Pressurizer Pressure is within normal operating range for operation at 100% Rated Thermal Power when PR-2B, Pressurizer PORV, fails 40% OPEN.

What are the Immediate Operator Actions in the correct order of performance AND the expected PRT pressure response of PR-2B failing OPEN 40%? (Immediate Operator Actions are completed within 1 minute)

- a. 1. CHECK Pressurizer Pressure < 2315 psig
2. CLOSE PR-2B
3. CLOSE Pressurizer Block Valve PR-1B

AND

PRT pressure will remain less than 200 psig due to the quench volume of the PRT.

- b. 1. CHECK Pressurizer Pressure < 2335 psig
2. CLOSE Pressurizer Block Valve PR-1B
3. CLOSE PR-2B

AND

PRT pressure will equalize with containment pressure due to the PRT rupture disk rupturing when PRT pressure reaches 120 psig.

- c. 1. CHECK Pressurizer Pressure < 2335 psig
2. CLOSE PR-2B
3. CLOSE Pressurizer Block Valve PR-1B

AND

PRT pressure will equalize with containment pressure when the piping rupture disk ruptures at 234 psig.

- d. 1. CHECK Pressurizer Pressure < 2315 psig
2. CLOSE Pressurizer Block Valve PR-1B
3. CLOSE PR-2B

AND

PRT pressure will increase consistent with the volume discharged from the Pressurizer.

ANSWER

d.

QUESTION # 035

Given the following:

- The unit was operating at 100% Rated Thermal Power when a reactor trip and Safety Injection occurred.
- Bkr 15201, Bus 52 Supply Bkr, OPENED on the reactor trip
- Diesel Generator 'A' failed to start.

With NO Operator action, which of the following equipment conditions is correct?

<u>Running</u>	<u>NOT Running</u>
a. ASV Exhaust Fan 'B'	RHR Pump 'A'
b. Component Cooling pump 'A'	Charging Pump 'B'
c. Safety Injection pump 'B'	Turbine Driven AFW Pump
d. Service Water Pump 1A1	CRPA Recirculation Fan 'B'

ANSWER

b.

QUESTION # 036

Given the following:

- The unit is at 100% Rated Thermal Power.
- The following indications are present in the Component Cooling (CC) Water System:
 - CC Pump 'A' - running
 - CC Pump 'B' - standby
 - CC return flow from the RXCP 'A' - 245 gpm and stable
 - Radiation Monitor R-17, Component Cooling, in ALERT
 - CC Surge tank level - 55% and slowly rising
 - CC header pressure - 32 psig and slowly lowering

What is an expected plant response for the current conditions?

- a. The running Component Cooling Pump, CC Pump 'A', receives a TRIP signal.
- b. The standby Component Cooling Pump, CC Pump 'B', receives an auto START signal.
- c. CC-610A, RXCP Thermal Barrier CC Isolation Valve, receives an automatic signal to CLOSE.
- d. CC-104, Component Cooling Surge Tank Vent Valve receives an automatic signal to CLOSE.

ANSWER

b.

QUESTION # 037

Given the following:

- The unit was operating at 60% power as indicated on Nuclear Instruments (NI), and stable.
- All Pressurizer Backup Heaters are in AUTO
- Charging Pump 'A' is operating in AUTO
- Charging Pump 'B' is operating in MANUAL
- A transient occurred with the following results:

<u>Parameter</u>	<u>Value</u>	<u>Trend</u>
Reactor Power (NI)	60%	Stable
Pressurizer Level	48%	Rising Slowly
Tave	562°F	Rising Slowly
Pressurizer Pressure	2263 psig	Rising Slowly

Which of the following combinations of listed pressurizer heaters and spray is the correct response for the current status?

Pressurizer Spray Valves
PS-1A & PS-1B

Backup Heaters

Heater Group 'C'

- | | | |
|--------------|-----|-----------|
| a. CLOSED | ON | MAX POWER |
| b. THROTTLED | ON | MIN POWER |
| c. CLOSED | OFF | MIN POWER |
| d. THROTTLED | OFF | MAX POWER |

ANSWER

b.

QUESTION # 038

Given the following:

- The unit is operating at 100% Rated Thermal Power.
- A total loss of main feedwater occurs.
- A reactor trip signal is generated, but the Reactor Trip Breakers FAIL to OPEN.
- A Rod Control Urgent Failure prevents rod motion.
- Auxiliary Feedwater System is operating per design.
- Operators have been dispatched to locally open the Reactor Trip Breakers.

What immediate action should be taken to mitigate this transient AND what is the purpose of the immediate action?

<u>Immediate Action</u>	<u>Purpose</u>
a. Align maximum Auxiliary Feedwater flow to ONE Steam Generator	Preserve a heat sink for RCS heat removal
b. Manually trip the Main Turbine	Preserve the water inventory in the Steam Generators
c. Open the Pressurizer PORVs	Avoid an RCS over-pressurization as the Pressurizer goes water solid
d. Slowly reduce turbine load	Prevent the PRZR PORVs from lifting by avoiding a rapid RCS temperature and pressure rise

ANSWER

b.

QUESTION # 039

Given the following:

- The unit was at 100% Rated Thermal Power.
- An equipment failure causes an inadvertent Safety Injection (SI) Signal.
- The operator determines the SI Signal is invalid and depresses the SI RESET pushbuttons 15 seconds after the SI initiation.

Which of the following is the result of the above operator actions?

- a. The running charging pumps will remain operating.
- b. SI RESET pushbuttons will need to be depressed again to reset SI.
- c. Containment Isolation will NOT occur due to resetting the SI signal.
- d. ALL components normally started by the SI signal will NOT automatically start.

ANSWER

b.

QUESTION # 040

Given the following:

- The Unit was operating at 100% Rated Thermal Power when the unit experienced a spurious reactor trip.
- During the resulting transient a Bus 3 and Bus 5 LOCKOUT occurred.

With NO operator action, which of the following lists equipment that would be RUNNING two (2) minutes after the transient?

- a. RXCP 'B', Containment Fan Coil Unit 'D', & Reactor Make Up Pump 'B'
- b. Rod Drive MG Set 1B, Main Feed Pump 'A', & Heater Drain Pump 'A'
- c. Boric Acid Transfer Pump 'B', Rod Drive MG Set 1A, & Fire Pump 'B'
- d. Containment Fan Coil Unit 'B', Containment Dome Fan 'B', & Condensate Pump 'B'

ANSWER

a.

QUESTION # 041

Given the following:

- 12:00:00 – The unit is in MODE 1 with the Shroud Cooling Coil 'A/B' in service
- 12:00:02 – Safety Injection (SI) occurs due to a small break LOCA
- 12:00:04 – The power supply to the Train 'A' Service Water (SW) Valves for Shroud Cooling fails, no other components are affected by the power loss.

Which of the following describes the expected valve response related to SW flow to Containment Fan Coil Unit (CFCU) 'B' and the effect on the given containment parameter?

'Shroud Cooling Coil A/B' SW valves (1) and the (2)

- a. (1) are unaffected by the SI Signal
(2) containment pressure rate of rise will be higher than expected for the stated accident.
- b. (1) reposition to isolate SW flow to Shroud Cooling Coil 'A/B'
(2) containment pressure rate of rise will be the same as if there was no loss of the stated power supply.
- c. (1) reposition to LIMIT the combined SW flow through Shroud Cooling Coil 'A/B' and CFCU 'B' to 50 gpm
(2) containment temperature rate of rise will be higher than expected for the stated accident.
- d. (1) remain in the same position with a CONSTANT SW flow rate is maintained to Shroud Cooling Coil 'A/B' and CFCU 'B'
(2) containment temperature rate of rise will be the same as if there was no loss of the stated power supply.

ANSWER

b.

QUESTION # 042

Given the following:

- The unit is in MODE 3.
- SP-23-100A, "Train A Containment Spray Pump and Valve Test – IST", is in progress.
- After starting Internal Containment Spray (ICS) Pump 'A', but before aligning ICS train 'A' for the full flow test, a Design Basis LOCA occurs.
- The crew enters the EOP network.
- Containment pressure is 25 psig and rising at the completion of immediate actions.

What is the status of the Internal Containment Spray Train 'A'? (SP-23-100A is provided as a reference)

ICS pump 'A' is providing . . .

- a. flow to ICS ring header 'A'.
- b. recirculation flow to the Refueling Water Storage Tank.
- c. no flow due to load shed and will start on the sequencer.
- d. no flow through the system and operating at shutoff head.

ANSWER

d.

QUESTION # 043

Given the following:

- The unit was operating at 100% Rated Thermal Power when a leak in the Main Steam System occurred.
- Annunciator 47052-F, STM Exclusion Area Temp High, is LIT.
- Annunciator 47051-F, STM Exclusion Area Isolation, is LIT.

Which of the following is an expected plant effect?

- a. Zone SV boundary dampers close and Zone SV Exhaust fans start.
- b. SI Active Status Panel Light 44901-0901, Zone SV EXH Fan A, is dim.
- c. Control Room A/C Fans and Auxiliary Building Supply Air Vent Fans trip.
- d. Zone SV Exhaust fan discharge dampers close and Zone SV Exhaust fans trip.

ANSWER

a.

44. Given the following:

- The unit is at 85% Rated Thermal Power.
- A Feedwater (FW) control malfunction results in the following:
 - S/G A level: 39% and stable
 - S/G B level: 72% and rising

Which of the following is a result of the above conditions?

- a. Only FW Pump 'A' is OPERATING
AND
FW Isolation Valve, FW-12B is CLOSED
- b. BOTH FW Pumps are OPERATING
AND
FW Regulating Valve, FW-7B is CLOSED
- c. Both FW Pumps are TRIPPED
AND
FW Isolation Valve, FW-12B is CLOSED
- d. FW Regulating Valve FW-7B is CLOSED
AND
BOTH FW Isolation Valves FW-12A/B are CLOSED

ANSWER

c.

45. Given the following:

- A reactor trip occurred from 100% Rated Thermal Power.
- All Auxiliary Feedwater (AFW) Pumps auto-started per design.
- A subsequent failure results in the loss of BRA-104, 125 VDC Bus, 30 seconds after Generator #1 Main Breaker G-1 OPENS.

Which of the following valve operations can successfully be performed from the Control Room?

- a. MS-102, Turbine Driven AFW Pump Main Steam Isolation, can be used to stop the Turbine Driven AFW pump.
- b. AFW-10A, AFW Train 'A' Crossover Valve, can be used to throttle discharge flow from the Turbine Driven AFW Pump.
- c. SW-502, Service Water to Turbine Driven AFW Pump Isolation, can be aligned to the suction of the Turbine Driven AFW Pump.
- d. AFW-10B, AFW Train 'B' Crossover Valve, can be used to isolate the Turbine Driven AFW Pump discharge flow to Steam Generator 'B'.

ANSWER

d.

QUESTION # 046

Given the following: (Events Occur In Order Listed)

- A loss of Off-Site power occurred.
- Neither Diesel Generator (DG) automatically started.
- A reactor trip and Safety Injection then occurred.
- DG 'A' was started as directed by ECA 0.0, "Loss of All AC Power".
- DG 'A' is running and carrying load.
- DG 'A' load indicates 2900 KW.

Which of the following correctly describes the status of DG 'A'?

- a. DG 'A' is within normal operational limits and may run continuously.
- b. DG 'A' is rated for up to 2000 hours of operation at the current load.
- c. Remove non-essential loads as necessary to reduce DG 'A' loading.
- d. Additional loads on DG 'A' may be started as necessary up to 2979 KW.

ANSWER

c.

QUESTION # 047

Given the following:

- The unit was operating at 100% Rated Thermal Power.
- A Station Blackout occurred.
- Station Battery 'A', BRA-101, is currently supplying plant DC loads.
- Discharge rate of Station Battery 'A', BRA-101, has been stable at 221 amp hours since the start of the Station Blackout.
- All station batteries were at their design capacity prior to the Station Blackout.

Which of the following is the operating design time of Station Battery 'A' at the current discharge rate?

- a. 2 hours
- b. 3 hours
- c. 4 hours
- d. 8 hours

ANSWER

d.

QUESTION # 048

Given the following:

- Annunciator 47102-D, Instrument Bus Inverter Trouble, is LIT.
- SER printout identifies BRA-112 as the affected inverter.

With NO other faults, which of the following is a correct indication based on the above conditions?

If BRA-105 is supplying the inverter's load . . .

- a. the inverter supplying load light would be ON.
- b. the alternate source supplying load light would be ON.
- c. annunciator 47101-A, BRA-102 DC Voltage Low, would be ON.
- d. the red circuit status light on BRA-104 for BRA-112 supply breaker would be OFF.

ANSWER

b.

QUESTION # 049

Given the following:

- Diesel Generator (DG) 'A' was started and is being prepared for parallel operations to the grid.
- See attached DG 'A' indications.

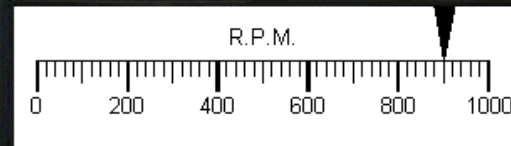
What will occur if Bkr 1-509, DG 'A' Output Breaker, closes under these conditions?

- a. DG 'A' will PICK UP LOAD and the synchronizing lights will be OFF.
- b. DG 'A' output CURRENT and SPEED will stabilize at a HIGHER value.
- c. Bkr 1-509 will TRIP on reverse power and DG 'A' voltage will LOWER.
- d. DG 'A' output voltage will RISE 4-6 volts and synchronizing lights will be ON.

ANSWER

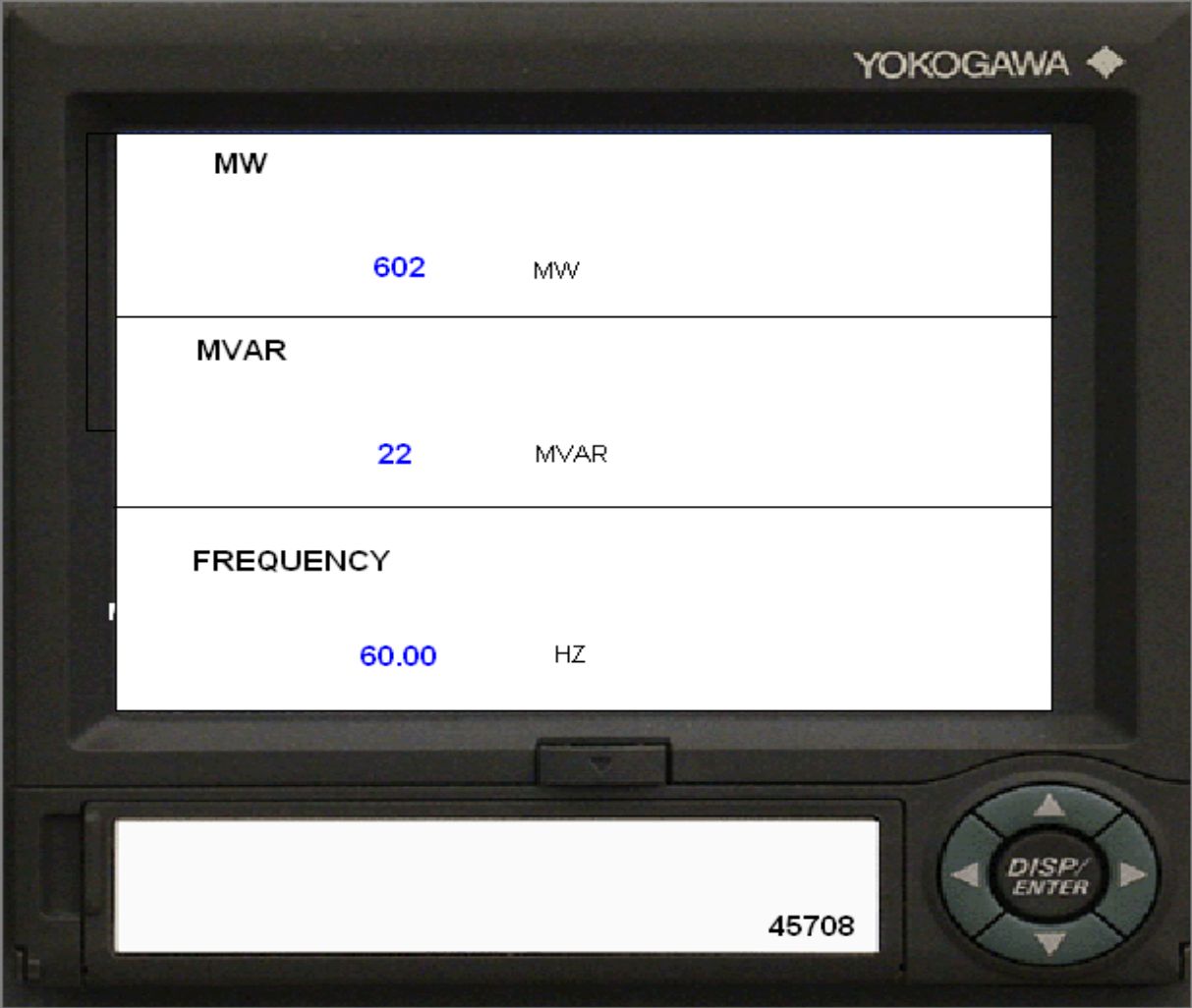
a.

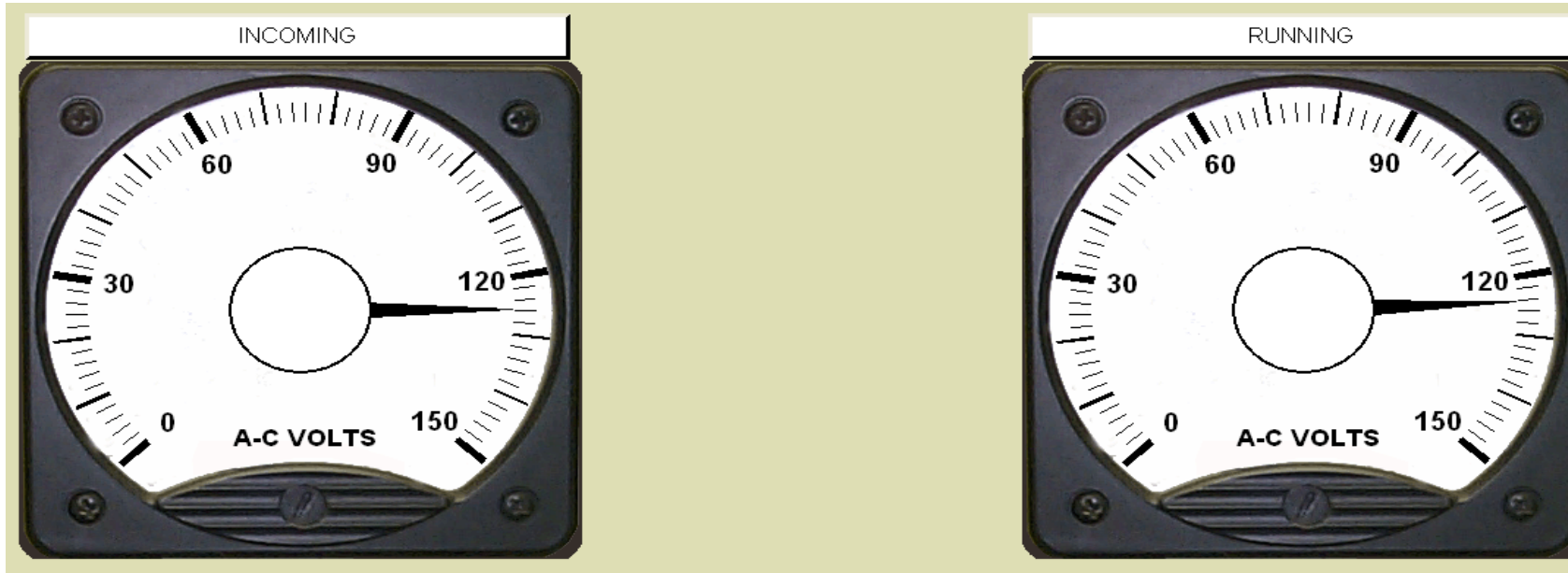
DIESEL GENERATOR A



MAIN GENERATOR

POWER





QUESTION # 050

Given the following:

- N-RM-45, "Radiation Monitoring System", is being performed to shutdown R-19, Steam Generator (SG) Blowdown Liquid Monitor.
- The Keyswitch for R-19 is positioned from ON to OFF.

Which of the following will result from this switch manipulation?

- a. Annunciator 47015-B, RAD Monitor Power Supply Failure, actuates due to removing power to R-19.
- b. Annunciator 47033-35, TLA-15 RMS Above Normal, actuates due to losing the communication link from R-19.
- c. A High Radiation (fail) signal is generated, and the SG Blowdown and SG Sample Isolation Valves automatically close.
- d. A High Radiation signal is defeated, preventing automatic closure of the SG Blowdown and SG Sample Isolation Valves.

ANSWER

c.

QUESTION # 051

Given the following:

- The unit is at 8% NI power, performing a startup after a refueling outage.
- Preparations to begin rolling the Main Turbine are in progress.
- The Balance Of Plant operator identifies that P-486, Turbine Impulse Pressure Instrument, is reading off scale high.
- Shortly after the failure of P-486, RXCP A Trips.
- Annunciator 47012-J, RCS Loop A Flow Low, is LIT.

What is the expected Plant AND Operator Response?

<u>Plant Response</u>	<u>Operator Response</u>
a. Permissive P-7 will not cause a reactor to trip	The operators will manually trip the reactor
b. Permissive P-8 will cause a reactor trip	The operators will perform the immediate actions for a reactor trip
c. Permissive P-10 will not cause a reactor trip	The operators will perform a normal reactor shutdown
d. Permissive P-13 will cause a reactor trip	The operators will manually trip the turbine

ANSWER

a.

QUESTION # 052

Given the following:

- The plant is operating at 100% Rated Thermal Power.
- Turbine Building Service Water Header Selector switch is in the 1A position.
- SW Header 'A' Pressure, PI-41503, indicates 58 psig and stable.
- SW Header 'B' Pressure, PI-41506, indicates 75 psig and stable.
- Forebay level is 50% and stable.

What procedure entry requirements are met?

- a. The entry conditions for AOP-GEN-001, "Operator Immediate Actions", are met and the crew shall perform the required immediate action.
- b. The entry conditions for AOP-SW-001, "Abnormal Service Water System Operation are met and the crew shall perform the procedure.
- c. Annunciator 47051-Q, Turbine Building Service Water Isolation, should be LIT and the crew shall perform the actions of the ARP.
- d. Annunciator 47052-M, Forebay Level Low, should be LIT and the crew shall perform the actions of the ARP.

ANSWER

b.

QUESTION # 053

Given the following:

- The unit is at 8% power by Nuclear Instruments following a refueling outage.
- IA-1303, IA to MS-1B Mn Stm Isol Valve 1B, was inadvertently CLOSED.
- A small air leak develops on one of the MSIV 'B' air accumulators.
- Annunciator 47063-K, MSIV AIR ACMTR Pressure Low, is LIT.

Which of the following is an expected response of plant?

- a. MS-1B, S/G B MSIV, will fail to CLOSE on a subsequent MSIV isolation signal.
- b. The alternate air accumulator will maintain MS-1B, S/G B MSIV, OPEN.
- c. A reactor trip will occur when MS-1B, S/G B MSIV, air pressure is LESS THAN 60 psig.
- d. A turbine trip will occur when MS-1B, S/G B MSIV comes 3° off the full OPEN position.

ANSWER

d.

QUESTION # 054

Given the following:

- The unit is operating at 100% Rated Thermal Power.
- Station and Instrument Air Compressor 'G' is OOS.
- Air Compressor 'C' is in OFF.
- Air Compressor 'F' is in service.
- Air Compressor 'F' trips on high HP air temperature.
- Instrument Air header pressure has lowered to 92 psig due to minor system leaks and is now stable.

Which of the following is the correct status for the equipment listed?

- a. Air Compressor 'B' is running and loaded.

AND

SA-400, Station Air Header 1B Isolation, is partially closed.

- b. Air Compressor 'B' is running and unloaded.

AND

SA-400, Station Air Header 1B Isolation, is fully open.

- c. Air Compressor 'A' is running and loaded.

AND

SA-400, Station Air Header 1B Isolation, is fully closed.

- d. Air Compressor 'A' is running and unloaded.

AND

SA-400, Station Air Header 1B Isolation, is modulating to maintain Instrument Air Header pressure > 92 psig.

ANSWER

a.

QUESTION # 055

Which containment personnel air lock malfunction affects BOTH the containment LCO AND the containment air locks LCO?

LCO 3.6.1 Containment shall be OPERABLE.

LCO 3.6.2 Two containment air locks shall be OPERABLE.

- a. Failure of the air lock interlock mechanism.
- b. Obstruction prevents closing the outer air lock door.
- c. Inability to lock containment air lock doors when they are closed.
- d. Total air lock leakage of 1.1 Containment Allowable Leakage Rate (L_a).

ANSWER

d.

QUESTION # 056

Which of the following lists a criteria that the 400 ft³ steam volume and 600 ft³ water volume of the Pressurizer are designed to meet?

- a. Keep the Pressurizer heaters covered during a 10% step load increase.
- b. Prevent a reactor trip due to Pressurizer level with a 50% ramped load reduction and Steam Dumps NOT available.
- c. During a reactor trip Pressurizer level will remain greater than 21% assuming steam dumps and/or Steam Generator relief valves lift as designed.
- d. The pressure surge from a 30% ramped load reduction with only automatic Rod Control operation will not exceed the design capacity of both Pressurizer Safeties.

ANSWER

a.

QUESTION # 057

Given the following:

- The reactor is initially critical below the Point of Adding Heat.
- Both Reactor Coolant Pumps are running.
- Main Steam Isolation Valves (MSIV) and MSIV bypass valves are closed.
- Moderator Temperature Coefficient is negative.
- RCS Tave is 547°F and stable with heat removal through the Steam Generator (SG) PORVs
- Intermediate Range (IR) Nuclear Instruments N-35 and N-36 indicate $1 \times 10^{-4}\%$ power.

If both SG PORVs fail CLOSED and NO operator action is taken, how will reactor power respond?

Reactor power will lower . . .

- a. temporarily, then return to $1 \times 10^{-4}\%$ power on Intermediate Range instruments due to subcritical multiplication.
- b. to a subcritical power level and stabilize at a value in the source range based on subcritical multiplication.
- c. initially, then return to $1 \times 10^{-4}\%$ power on Intermediate Range instruments due to the resulting rise in RCS Tave.
- d. until the positive reactivity from the temperature rise causes the reactor to stabilize at a critical power level in the Source Range.

ANSWER

b.

QUESTION # 058

Given the following:

- The unit has been operating at 100% Rated Thermal Power for the last three months.
- The following annunciators are LIT:
 - 47033-12, TLA-2 RCS Subcooling High/Low
 - 47033-24, TLA-9 Core Exit T/C Tilt
 - 47044-F, ICCMS Panel Trouble

Which of the following plant conditions could cause ALL of the above alarms?

- a. Train 'A' of Reactor Vessel Level Indication fails to 0%.
- b. RCS Wide Range Pressure Channel PT-420 fails low.
- c. One Core Exit Thermocouple is 671°F and rising slowly.
- d. A momentary reactor power excursion causes a spike to 102%.

ANSWER

c.

QUESTION # 059

Which of the following correctly describes the combination of systems that are used to control Containment hydrogen concentration following a Design Basis LOCA?

- a. Waste Gas system

AND

Containment Ventilation System

- b. Post LOCA Train 'A'

AND

Containment Vacuum Breakers

- c. Instrument Air System

AND

Shield Building Ventilation System

- d. Containment Dome Vent Fans

AND

Auxiliary Building Ventilation System

ANSWER

c.

QUESTION # 060

Given the Following

- The unit is in MODE 6.
- Core Off-Load has NOT begun.
- Reactor Cavity water level = 23 ft above the top of the reactor vessel flange.
- Spent Fuel Pool (SFP) Pump 'A' is isolated for maintenance.
- 'B' Residual Heat Removal (RHR) Heat Exchanger is isolated due to a tube leak.
- A breaker fault caused a loss of power to the bus supplying power to SFP Pump 'B'; the bus is NOT locked out.
- Annunciator 47055-N, Spent Fuel Pool Abnormal, is LIT.
 - SER 157, Spent Fuel Pool A Temperature High, is in ALARM.
 - SER 158, Spent Fuel Pool B Temperature High, is in ALARM.
- Local SFP Temperatures are 105°F and slowly rising.

For the given indications, which of the following actions would you expect to be directed per AOP-SFP-001, "Abnormal Spent Fuel Pool Cooling and Cleanup System Operation"?

- a. Close the crosstie Bkr from Bus 46 to provide power to SFP Pump 'B'.
- b. Align SFP Cooling to the RHR 'A' Heat Exchanger to provide cooling to the SFP.
- c. Ensure Auxiliary Building Ventilation isolation is activated to minimize release to the Auxiliary Building.
- d. Add makeup to the SFP to maintain SFP level between low level alarm and high level alarm to maintain boil off inventory.

ANSWER

d.

QUESTION # 061

Given the following:

- The unit has been at 100% Rated Thermal Power for 100 days.
- Fuel is being moved in the Spent Fuel Pool (SFP) North Pool in preparation to be loaded into a Dry Cask.
- The SFP canal is pumped down to support maintenance on the upender pivot pins.

If the South Pool Transfer Canal gate seal completely fails, and there are NO operator actions taken, what is the expected final level of the SFP?

SFP level will be approximately ____ feet below the Spent Fuel Pool floor.

- a. 2
- b. 4
- c. 9
- d. 17

ANSWER

c.

QUESTION # 062

Given the following:

- The unit is at 100% Rated Thermal Power.
- Electrical systems are in a normal lineup when Main Transformer 'B' experiences a Sudden Pressure.
- Annunciator 47075-I, MAIN XFRM Sudden Pressure, is LIT.

Which of the following describes a plant effect of the above events?

- a. Generator #1 Field Breaker trips causing Generator #1 Main Breaker G-1 to open
AND

A fast transfer of Buses 1-1, 1-2, 1-3, 1-4 and 1-5 to the Reserve Auxiliary Transformer

- b. Generator #1 Main Breaker G-1 opens causing Diesel Generators 1A and 1B to start

AND

A reactor trip causes a subsequent turbine trip

- c. Associated 63X Sudden Pressure Relays actuate and Diesel Generators 1A and 1B start

AND

Buses 1-5 and 1-6 transfer and load to their respective Diesel Generators

- d. 86/T1A and 86/T1B Main Generator lockout relays actuate and Generator #1 Field Breaker trips

AND

A turbine trip signal is generated

ANSWER

d.

63. Given the following:

- A radioactive waste release of the Gas Decay Tank 'A' to the Auxiliary Building Vent is in progress when the following annunciators alarm:
 - TLA-15, RMS Above Normal
 - 47011-B, Radiation Indication High
 - 47012-B, Radiation Indication Alert
 - 47045-M, Waste Disposal Abnormal
- The HIGH alarm is LIT for the following Radiation Monitors
 - R-13, Aux Bldg Vent Exhaust Radiation Monitor, HIGH light is LIT
 - R-14, Aux Bldg Vent Exhaust Radiation Monitor, HIGH light is LIT.
- The NAO reports the release of Gas Decay Tank 'A' was automatically terminated.

What automatically terminated the release of Gas Decay Tank 'A' to the Auxiliary Building Ventilation System?

- a. Waste Gas Compressor 'A' TRIPPED.
- b. WG-36, Gaseous Waste Discharge Valve, CLOSED automatically.
- c. WG-34A, Gas Decay Tank A Outlet Valve, CLOSED automatically.
- d. The discharge is automatically realigned to the Dearated Drain Tank.

ANSWER

b.

QUESTION # 064

Which of the following maintenance activities would affect the status of LCO 3.7.8.?

LCO 3.7.8 - Two Service Water trains shall be OPERABLE.

- a. Electrical Maintenance troubleshooting results in a loss of the indication of Service Water Pump amps.
- b. I&C miscalibration results in the Circ. Water pump NOT tripping at the required Forebay Water level.
- c. One train of Circ. Water Flooding Protection Trip Circuitry is rendered nonfunctional during an I&C Surveillance.
- d. SW-1300B, Comp Cooling Heat Exchanger B Outlet, is identified closed during a maintenance tagout walkdown.

ANSWER

b.

QUESTION # 065

Which of the following functions are required for the following LCO?

LCO 3.3.4 The Dedicated Shutdown System Functions shall be OPERABLE.

- a. RCS Pressure Control
 - Decay Heat Removal via the AFW system
 - On-Site Power to support RCS inventory control
- b. Technical Support Center Habitability
 - Decay Heat Removal via the Steam Dumps
 - Emergency DC power to support Instrumentation
- c. Spent Fuel Pool Cooling
 - Safety Injection via the Safety Injection Pumps.
 - Service Water to support Emergency Diesel Generator operation
- d. Long Term Reactivity Control
 - Decay Heat Removal via the Steam Generator PORVs
 - Component Cooling Water to support Containment Spray Pump operation

ANSWER

a.

QUESTION # 066

Given the following:

- The unit was operating at 100% Rated Thermal Power when the reactor tripped.
- The crew has completed the required steps of E-0, "Reactor Trip or Safety Injection" and transitioned to ES-0.1, "Reactor Trip Response"
- Main Steam Isolation Valves are OPEN
- The Balance of Plant Operator reports:
 - MS-312A-1, Gland Seal Steam to MSR B1 & B2 Relief Vlvs is OPEN
 - MS-312B-1, Gland Seal Steam to MSR B1 & B2 Relief Vlvs is CLOSED

What actions will be required to mitigate this?

- a. Condenser steam dump valves will be used for controlling RCS temperature.
- b. 'A' Train Condenser Steam Dump valves are still available for RCS temperature control.
- c. Main Steam Isolation Valves will be closed to reduce the excessive cooldown of RCS temperature.
- d. Steam Generator PORVs or Atmospheric Steam Dump valves will be used to control RCS temperature.

ANSWER

d.

QUESTION # 067

Which of the following lists the correct application of Continuous Action Steps (CAS) during the performance of Emergency Operating Procedures?

- a. Does NOT apply after a transition to another Emergency Operating Procedure.
- b. Applies until the action stated in the step has been completed after which its applicability ends.
- c. Applies to the entire procedure and its applicability continues when transitioning to a Normal Operating Procedure.
- d. Applies to the current procedure only after the step is first procedurally encountered, and may apply after a transition is made to another Emergency Operating Procedure.

ANSWER

d.

QUESTION # 068

Which of the following lists components that can ALL be operated from the Dedicated Shutdown Panel?

- a. Fuel Oil Transfer Pump 'A'
CVC-15, CHRG Line To PRZR Aux Spray
MS-1A, Main Steam HDR 1A Isolation Valve
- b. Charging Pump 'A'
Battery Room Fan Coil Unit 'A'
LD-13A, Reactor Coolant Filter Bypass Valve
- c. RHR pump 'A'
BT-3A, STM GEN 1A Blowdown Isol
LD-4B, Regen HX LTDN OTLT ORIF 1B Isol
- d. Containment Fan Coil Unit 'B'
Service Water Pump 1B1
RHR-11, RHR Return to Loop B Cold Leg Isol

ANSWER

a.

QUESTION # 069

Given the following:

- The unit is operating at 100% Rated Thermal Power when both SI Accumulators are declared INOPERABLE per LCO 3.5.1
- The Unit Supervisor enters LCO 3.0.3.

LCO 3.5.1 Two SI accumulators shall be OPERABLE

Which of the following combinations of Reactor Coolant System Temperature, and Pressure allows exiting LCO 3.0.3. in the SHORTEST amount of Time?

(Assume that the SI Accumulators remain INOPERABLE)

	<u>RCS Temperature</u>	<u>RCS Pressure</u>	<u>Time Since INOPERABILITY</u>
a.	199°F	300 psig	34 hours
b.	350°F	400 psig	12 hours
c.	450°F	600 psig	6 hours
d.	540°F	2235 psig	5 hours

ANSWER

c.

QUESTION # 070

Which of the following is an example of acceptable preconditioning before performing a Technical Specification surveillance?

- a. Removing electrical loads prior to a load rejection test.
- b. Lubricating valve stems prior to as found stroke time testing.
- c. Performing operational readiness checks prior to performance of the surveillance.
- d. Performance of maintenance activities prior to a surveillance test with the intent of ensuring favorable test results.

ANSWER

c.

QUESTION # 071

Why is it preferred that Post Accident Leakage Control System be activated per AOP-MDS-002, "Post Accident Leakage Control System", prior to establishing containment sump recirculation during a Large Break LOCA?

- a. To prevent boron dilution of Containment Sump 'B' during containment sump recirculation.
- b. The Post Accident Leakage Control System is required to obtain containment sump samples during containment sump recirculation.
- c. High radiation levels could prohibit diverting the Deaerator Drain Tank Vent to containment during containment sump recirculation.
- d. A large differential pressure across CVC-215B, Seal Water Filter Bypass valve, will develop during containment sump recirculation and prevent it from opening.

ANSWER

c.

QUESTION # 072

What are the established Dominion Nuclear Total Effective Dose Equivalent Administrative Limits for Radiation Workers?

- a. At the home site – 125 mrem per year
All Licensees or utilities – 500 mrem per year
- b. At the home site – 500 mrem per year
All Licensees or utilities – 1,000 mrem per year
- c. At the home site – 2,000 mrem per year
All Licensees or utilities – 3,000 mrem per year
- d. At the home site – 4,500 mrem per year
All Licensees or utilities – 5,000 mrem per year

ANSWER

c.

QUESTION # 073

For which of the following tasks is a General RWP allowed?

- a. To control work in areas having dose rates > 200 mrem/hour at 30 cm.
- b. Entry into a Locked High Radiation Area or Very High Radiation areas for the sole purpose of performing a plant inspection.
- c. Plant Inspections in High Radiation Areas when the radiological conditions are static and approved by the Manager of Radiological Protection and Chemistry.
- d. Plant inspections where the expected dose exceeds 100 mrem per individual per entry when approved by the Manager of Radiological Protection and Chemistry.

ANSWER

c.

QUESTION # 074

You are a licensed Reactor Operator working on outage tagouts in the Work Control Center; this is your relief week. The following plant announcement is made:

“Attention all personnel, Attention all personnel. We are experiencing an ALERT. Emergency Response Organization personnel should report to their duty locations. All other personnel should report to the nearest assembly area.”

To which of the following locations do you report?

- a. The Control Room
- b. Site Relocation Facility
- c. Admin Assembly Room
- d. Warehouse Annex Lunch Room

ANSWER

a.

QUESTION # 075

Given the following:

- The unit was operating at 100% Rated Thermal Power when a fault caused a failure of the Reserve Auxiliary Transformer.
- The following annunciators are LIT after the automatic response of the plant; NO operator actions have been taken.
 - 47013-A, Rad Monitor Sampling Flow High/Low
 - 47041-J, Charging Pump Trip
 - TLA-11, Reactor Thermal Power High
 - 47062-M, AFW Pump B Abnormal
 - 47082-E, RAT Lockout
 - 47082-G, Rat Differential Current
 - 47084-A, Substation Major

Based on the given annunciators in alarm, which of the following actions has priority?

- a. Stop AFW Pump 'B'
- b. Establish charging flow
- c. VPL down to lower power
- d. Align R-21 to sample containment

ANSWER

c.

QUESTION # 076

Given the following:

- A tube rupture in Steam Generator (SG) 'A' has been diagnosed.
- The crew is performing the actions of E-3, "Steam Generator Tube Rupture".
- RCS pressure and SG 'A' pressure have been stabilized at 850 psig.
- Subcooling is 65°F.
- Pressurizer level is being maintained at 54% with charging and letdown in service.
- SG 'A' narrow range level is 68% and stable.
- SG 'B' narrow range level is 10% and stable.
- CST levels are at 35% and makeup to the CSTs CANNOT be established.
- Management has directed that the cooldown and depressurization be conducted as quickly as possible.

What procedure will be used in this condition?

- a. ECA-3.2, "SGTR With Loss of Reactor Coolant – Saturated Recovery Desired".
- b. ES-3.1, "Post-SGTR Cooldown Using Backfill".
- c. ES-3.2, "Post-SGTR Cooldown Using Blowdown".
- d. ES-3.3, "Post-SGTR Cooldown Using Steam Dump".

ANSWER

d.

QUESTION # 077

According to the Applicable Safety Analysis section of Technical Specifications, which of the listed LCOs directly mitigates the adverse consequences of a Main Steam Line Break?

- a. LCO 3.7.4 Two Steam Generator Power Operated Relief Valves lines shall be OPERABLE
- b. LCO 3.7.3 Two Main Feedwater Isolation Valves, two Main Feedwater Regulating Valves and Main Feedwater Regulating Valves Bypass Valves shall be OPERABLE
- c. LCO 3.3.2 Engineered Safety Feature Actuation System instrumentation for each Function in TABLE 3.3.2-1 shall be OPERABLE:
 - 1. Function 3.a, Containment Isolation – Manual Initiation
- d. LCO 3.3.2 Engineered Safety Feature Actuation System instrumentation for each Function in TABLE 3.3.2-1 shall be OPERABLE:
 - 1. Function 5.b Feedwater Isolation – Steam Generator Water Level High-High

ANSWER

b.

QUESTION # 078

Given the following:

- The unit was operating at 100% Rated Thermal Power when the reactor tripped and Safety Injection (SI) initiated.
- The crew entered the EOP network and transitioned from E-0, "Reactor Trip or Safety Injection", to E-1, "Loss of Reactor or Secondary Coolant".
- Immediately after Resetting SI and Containment Isolation the site experienced a Loss of Off-Site Power.
- Parameters at the time of the Loss of Off-Site Power:

<u>Parameter</u>	<u>Value</u>	<u>Trend</u>
Containment Pressure	3 psig	Slowly Lowering
RCS Pressure	700 psig	Slowly Lowering
RWST Level	47%	Lowering
RCS Subcooling	0°F	Stable
Pressurizer Level	0%	-----
SG 'A' Narrow Range Levels	8%	Rising Uncontrollably
SG 'B' Narrow Range Levels	0%	-----

What actions will the Unit Supervisor direct?

- Manual starting of SI Pumps and isolation of feed flow to 'A' SG while continuing in E-1, "Loss of Reactor or Secondary Coolant".
- Transition to ES-0.0, "Rediagnosis". During the performance of ES-0.0 direct isolation of feed flow to SG 'A' and initiation of SI using the pushbuttons.
- Manual starting of SI Pumps then transition to E-3, "Steam Generator Tube Rupture". After the transition to E-3 direct isolation of feed flow to SG 'A'.
- Initiation of SI using the pushbuttons then transition to E-0, "Reactor Trip or Safety Injection", step 4. After the transition to E-0 direct isolation of feed flow to SG 'A'.

ANSWER

c.

QUESTION # 079

Given the following:

- The unit is at 100% Rated Thermal Power.
- A fault caused Bkr 15206 to open, de-energizing MCC 1-52B & MCC 1-52C.
- BRA-102, 125 VDC Bus, voltage is 125 VDC.
- The crew has verified power is available to the affected inverter(s) static switch(es).

DETERMINE the OPERABILITY status of Red Channel Instrument Bus I AND reason for the Operability Status?

OPERABILITY StatusReason

- | | |
|---------------|---|
| a. INOPERABLE | The fault has caused an INOPERABILITY of the Train 'A' safeguards battery which is a support system of the Red Channel Instrument Bus |
| b. INOPERABLE | The Red Channel Instrument Bus power is not supplied from an uninterruptible power source |
| c. OPERABLE | The fault has caused an INOPERABILITY of the Train 'A' safeguards battery which is not a support system of the Red Channel Instrument Bus |
| d. OPERABLE | The Red Channel Instrument Bus still has power |

ANSWER

d.

QUESTION # 080

Given the following:

- The unit is in MODE 6.
- A core re-load is in progress.
- Emergency Diesel Generator B is out of service for maintenance.
- RAT is out of service for maintenance.
- The following annunciators simultaneously alarm:
 - 47034-C, SFGRD A or B Control Power Failure
 - 47901-B, Diesel Generator A Mech Lockout
 - 47092-B, Diesel Gen A Local Control.
 - 47093-A, Diesel Gen A Control Volt Low
 - 47101-A, BRA-102 DC Voltage Low
 - 47102-A, BRA-102 Feeder Bkr Undervoltage
 - 47103-A, BRA-104 Feeder Bkr Undervoltage
- BRA-102 voltage is 0 VDC.
- The Unit Supervisor has entered AOP-EDC-002, "Loss of Train A Safeguards DC Power".
- The current step in AOP-EDC-002 requires an evaluation of TS 3.8.10 Distribution Systems – Shutdown.

Using the provided references, DETERMINE which condition requires entry?

- a. LCO 3.8.2 Condition A
- b. LCO 3.8.3 Condition A
- c. LCO 3.8.5 Condition A
- d. LCO 3.8.8 Condition A

ANSWER

c.

AC Sources - Shutdown
3.8.2

3.8 ELECTRICAL POWER SYSTEMS

3.8.2 AC Sources - Shutdown

LCO 3.8.2 The following AC electrical power sources shall be OPERABLE:

- a. One qualified circuit between the offsite transmission network and the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems - Shutdown"; and
- b. One diesel generator (DG) capable of supplying one train of the onsite Class 1E AC electrical power distribution subsystem(s) required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required offsite circuit inoperable.	-----NOTE----- Enter applicable Conditions and Required Actions of LCO 3.8.10, with one required train de-energized as a result of Condition A.	
	A.1 Declare affected required feature(s) with no offsite power available inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	

AC Sources - Shutdown
3.8.2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u> A.2.3 Initiate action to restore required offsite power circuit to OPERABLE status.	Immediately
B. One required DG inoperable.	B.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u> B.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u> B.3 Initiate action to restore required DG to OPERABLE status.	Immediately

Diesel Fuel Oil and Lube Oil
3.8.3

3.8 ELECTRICAL POWER SYSTEMS

3.8.3 Diesel Fuel Oil and Lube Oil

LCO 3.8.3 The stored diesel fuel oil and lube oil shall be within limits for each required diesel generator (DG).

APPLICABILITY: When associated DG is required to be OPERABLE.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each DG.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more DGs with usable combined storage and day tanks fuel level < 32,888 gal and > 27,961 gal.	A.1 Restore fuel oil level to within limits.	48 hours
B. One or more DGs with lube oil inventory < 504 gal and > 432 gal.	B.1 Restore lube oil inventory to within limits.	48 hours
C. One or more DGs with stored fuel oil total particulates not within limit.	C.1 Restore fuel oil total particulates to within limits.	7 days
D. One or more DGs with new fuel oil properties not within limits.	D.1 Restore stored fuel oil properties to within limits.	30 days

Diesel Fuel Oil and Lube Oil
3.8.3

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A, B, C, or D not met. <u>OR</u> One or more DGs with diesel fuel oil or lube oil not within limits for reasons other than Condition A, B, C, or D.	E.1 Declare associated DG inoperable.	Immediately

DC Sources - Shutdown
3.8.5

3.8 ELECTRICAL POWER SYSTEMS

3.8.5 DC Sources - Shutdown

LCO 3.8.5 One DC electrical power subsystem shall be OPERABLE to support one subsystem of the DC Electrical Power Distribution System required by LCO 3.8.10, "Distribution System - Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

NOTE

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required DC electrical power subsystem inoperable.	A.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore required DC electrical power subsystem to OPERABLE status.	Immediately

Inverters - Shutdown
3.8.8

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 Inverters - Shutdown

LCO 3.8.8 One inverter shall be OPERABLE to support the 120 VAC electrical distribution subsystem required by LCO 3.8.10, "Distribution Systems - Shutdown."

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

-----NOTE-----

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	A.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore required inverter to OPERABLE status.	Immediately

Distribution Systems - Shutdown
3.8.10

3.8 ELECTRICAL POWER SYSTEMS

3.8.10 Distribution Systems - Shutdown

LCO 3.8.10 The necessary portion of AC, DC, and AC instrument bus electrical power distribution subsystems shall be OPERABLE to support equipment required to be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

NOTE

LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC, DC, or AC instrument bus electrical power distribution subsystems inoperable.	A.1 Declare associated supported required feature(s) inoperable.	Immediately
	<u>OR</u>	
	A.2.1 Suspend movement of irradiated fuel assemblies.	Immediately
	<u>AND</u>	
	A.2.2 Suspend operations involving positive reactivity additions that could result in loss of required SDM or boron concentration.	Immediately
	<u>AND</u>	

Distribution Systems - Shutdown
3.8.10

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.3 Initiate actions to restore required AC, DC, and AC instrument bus electrical power distribution subsystems to OPERABLE status.	Immediately
	<u>AND</u> A.2.4 Declare associated required residual heat removal subsystem(s) inoperable and not in operation.	Immediately

QUESTION # 081

Given the following:

- The unit was operating at 100% Rated Thermal Power when the reactor tripped and Safety Injection initiated.
- The crew entered the EOP network and transitioned from E-1, "Loss of Reactor or Secondary Coolant" to ES-1.3, "Transfer to Containment Sump Recirculation".
- Upon Entry in ES-1.3 the Reactor Operator reports that Containment Wide Range Level is less than 2 feet and stable.

Which of the following procedural flow paths maintains adherence to and proper selection of the appropriate procedures for the given situation?

The Unit Supervisor will (1) and (2) .

- a. continue in ES-1.3, "Transfer To Containment Sump Recirculation"

if Containment Wide Range Level is < 2 feet when RWST Level is \leq 10% then transition ECA-1.2, "LOCA Outside of Containment"

- b. continue in ES-1.3, "Transfer To Containment Sump Recirculation"

if Containment Wide Range Level is < 2 feet when RWST Level is \leq 4% then transition ECA-1.1, "Loss of Containment Sump Recirculation"

- c. transition to ECA-1.1, "Loss of Containment Sump Recirculation"

if the RHR pumps begin cavitating and minimum RCS injection flow cannot be maintained then transition to ECA-1.3, "Containment Sump Blockage"

- d. transition to ECA-1.1, "Loss of Containment Sump Recirculation"

if the minimum RCS injection flow cannot be maintained then transition to FR-C.1, "Response to Inadequate Core Cooling" when CETs > 600°F

ANSWER

c.

QUESTION # 082

Which of the following identifies an accident that the RCS Specific Activity Technical Specification limits are based on, AND what action is requested during implementation of AOP-RC-003, "High Reactor Coolant Activity"?

<u>Accident</u>	<u>Action Requested</u>
a. Main Steam Line Break	Perform NF-AA-3004, "Fuel Integrity Monitoring"
b. Loss Of Coolant Accident	Perform OSP-RCS-001, "Reactor Coolant Leak Rate Check"
c. Steam Generator Tube Rupture	Perform EPIP-RET-03A, "Liquid Effluent Release Paths"
d. Faulted and Ruptured Steam Generator	Perform CY-KW-059-003, "Primary to Secondary Leak Rate Data"

ANSWER

a.

QUESTION # 073

Given the following:

- The unit is in MODE 6.
- Fuel is currently being offloaded from the core.
- NCL-FH-004, "Refueling Daily Checklist" has been completed.
- N-31 Source Range is selected and providing the audible count rate to the Control Room.
- N-32 Source Range audible count rate is NOT functional.

If the Control Room audible count rate provided by N-31 is lost, what action is required by Technical Specifications AND what is the basis for that action?

<u>Action</u>	<u>Basis</u>
a. Initiate actions to isolate unborated water sources	Without a direct method to monitor reactivity conditions in the Control Room suspend all actions with the potential of positive reactivity addition until the containment audible count rate is verified OPERABLE
b. Initiate actions to isolate unborated water sources	Prompt and definite indication of any boron dilution is no longer provided with the loss of the audible count rate instrumentation
c. Suspend all operations which will cause an RCS cooldown	Minimizing cooldown of the RCS will ensure positive, reactivity due to temperature changes will not be added
d. Suspend all operations which will cause an RCS cooldown	Without a direct method to monitor reactivity conditions in the Control Room suspend all actions with the potential of positive reactivity addition until the containment audible count rate is verified OPERABLE

ANSWER

b.

QUESTION # 084

Given the following:

- The unit is operating at 100% Rated Thermal Power.
- SD-2A, SG 'A' PORV Isolation Valve, closed due seat leakage of SD-3A, SG 'A' PORV.
- A common fault causes both Main Steam Isolation Valves to CLOSE.
- The reactor fails to trip automatically.
- All rods are inserted when the Operators successfully perform the RESPONSE NOT OBTAINED Actions for Step 1 of E-0, "Reactor Trip or Safety Injection".
- After the immediate actions of E-0 are completed security reports steam on the east side of containment and no visible steam on the west side of containment.
- Safety Injection has NOT automatically initiated.
- After the completion of the immediate actions of E-0, the following plant parameters are noted by the operators:

<u>Parameter</u>	<u>Value</u>	<u>Trend</u>
RCS Pressure	2340 psig	Lowering
PRZR Level	55%	Stable
Loop 'A' RCS Hot Leg Temp	576°F	Stable
Loop 'B' RCS Hot Leg Temp	576°F	Stable
Loop 'A' RCS Cold Leg Temp	575°F	Stable
Loop 'B' RCS Cold Leg Temp	552°F	Very Slowly Lowering
SG 'A' pressure	1200 psig	Stable
SG 'B' pressure	1060 psig	Very Slowly Lowering
Containment Pressure	0.1 psig	Stable
RCS Void Fraction	< 0%	Stable
Both SG Narrow Range Levels	> 5%	Slowly Increasing

What procedure will the Unit Supervisor transition to in order to mitigate the conditions?

- a. E-0, "Reactor Trip or Safety Injection"
- b. FR-C.2, "Response to Degraded Core Cooling"
- c. FR-C.3, "Response to Saturated Core Cooling"
- d. FR-H.2, "Response to Steam Generator Overpressure"

ANSWER

d.

QUESTION # 085

Given the following:

- The unit was operating at 100% Rated Thermal Power when a Large Break LOCA occurred
- A common fault caused the TAT and Bus 5 to Lockout when the reactor tripped
- The crew has entered the EOP network and is currently performing actions in E-1, "Loss of Reactor or Secondary Coolant"
- Containment Pressure is 17 psig and lowering very slowly
- SW Flow to CFCU 'A' 0 gpm
- SW Flow to CFCU 'B' 0 gpm
- SW Flow to CFCU 'C' 1600 gpm
- SW Flow to CFCU 'D' 900 gpm
- Containment Sump Wide Range Level is 8.0 ft and slowly trending up
- RWST Level is 50% and slowly lowering
- R-16, Containment Fan Cooling Unit Service Water Return, radiation monitor reads 350 cpm and trending up

Which procedure(s) will the Unit Supervisor direct?

- a. Transition to FR-Z.2, "Response to Containment Flooding"
- b. Continue in E-1 while concurrently implementing AOP-SW-001, "Abnormal Service Water System Operation"
- c. Transition to ECA-1.3, "Containment Sump Blockage", and perform ARP-47052-A, "Reactor Building Ventilation Abnormal", in parallel with ECA-1.3
- d. Perform AOP-EHV-007, "Loss of Off-Site Power", while monitoring for parameters that indicate a required transition to ES-1.3, "Containment Sump Recirculation"

ANSWER

a.

QUESTION # 086

Given the following:

- Unit restart has commenced 16 hours after a spurious reactor trip
- Reactor Power is at 4% of Rated Thermal Power
- Tave is 547°F being maintained by the steam dumps
- Pressurizer Pressure Control is in Automatic controlling pressure at 2235 psig
- PT-429, Red Channel Pressurizer Pressure, has FAILED HIGH
- The crew has implemented AOP-MISC-001, "Response to Instrument Failure"
- AOP-MISC-001 directs the Unit Supervisor to Refer to TS 3.3.1 and TS 3.3.2

LCO 3.3.1 The RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

Which instrumentation Function requires ENTRY into a LCO CONDITION AND what is the protection provided by this function?

<u>Function</u>	<u>Protection</u>
a. Low Pressure safety injection function.	Prevent exceeding RCS pressure safety limit.
b. Overtemperature Delta -T reactor trip function.	Prevent exceeding DNBR limits.
c. Pressurizer Low Pressure reactor trip function.	Prevent exceeding subcooling limits during MODE 1 operations.
d. Pressurizer High Pressure reactor trip function.	Prevent exceeding the peak centerline fuel temperature limit specified in TS.

ANSWER

b.

QUESTION # 087

Given the following:

- The plant was operating at 100% power when a Design Basis LOCA occurred.
- A loss of Off-Site power occurred when the reactor tripped.
- ATC reports that the expected return of Off-Site power is in 3 hours.
- EDG 'B' failed to start.
- Annunciator 47091-J, Bus 6 Lockout, is CLEAR.
- ICS Pump 'A' is tagged out for planned maintenance.
- All other ECCS equipment is functioning as expected.

What plant conditions are expected as a result of the above failures AND what procedure would be used to mitigate the failure of ICS?

<u>Plant Conditions</u>	<u>Procedure</u>
a. Containment temperature and pressure will rise above the auto ignition limits for hydrogen	NOP-RBV-003, "Post LOCA Hydrogen Control", actions will lower hydrogen concentration by directing actions to dilute the containment hydrogen concentration
b. Containment pressure will rise and there will be a potential loss of containment integrity	AOP-DGM-002B, "Abnormal Diesel Generator B Operation", actions will restore power to Bus 6 by directing actions to start EDG 'B'
c. Containment temperature will exceed design values for adverse instrumentation	FR-Z.1, "Response to High Containment Pressure", actions will reduce containment temperature and pressure by directing actions to vent containment to the atmosphere
d. Airborne radiation levels in containment will be elevated	AOP-RM-001, "Abnormal Radiation Monitoring", actions will reduce the release rate to the environment by directing actions to vent containment to the shield building

ANSWER

b.

QUESTION # 088

Given the following:

- The unit was operating at 100% Rated Thermal Power when the reactor automatically tripped and Safety Injection initiated.
- Off-Site power was lost at the same time the reactor trip breakers opened.
- Containment Pressure is 28 psig.
- Containment Radiation is 5.5×10^5 R/Hr.

Based on the stated containment conditions, which of the following correctly states when the Unit Supervisor can direct stopping of Internal Containment Spray Pumps and be in compliance with station procedures?

The Unit Supervisor can direct stopping . . .

- a. one ICS pump when containment pressure is ≤ 17 psig during the performance of E-1, "Loss of Reactor or Secondary Coolant".
- b. one ICS pump during the performance of ES-1.3, "Transfer to Containment Sump Recirculation", with current containment conditions.
- c. both ICS pumps during the performance of ECA-2.1, "Uncontrolled Depressurization of Both Steam Generators", when containment pressure lowers to ≤ 23 psig.
- d. both ICS pumps during the performance of ECA-3.1, "SGTR With Loss of Reactor Coolant Subcooled Recovery Desired", when containment radiation levels lowers to $\leq 5.0 \times 10^5$ R/Hr.

ANSWER

b.

QUESTION # 089

Given the following:

- During the performance of OSP-DGE-002A, "Diesel Generator Quarterly Availability Test", SW-301A, Service Water Supply to A Diesel Generator, failed to OPEN automatically.
- An operator opened SW-301A locally upon Control Room direction.

Which of the following describes the status of LCO 3.8.1 & LCO 3.7.8?

- LCO 3.8.1 The following AC electrical sources shall be OPERABLE:
- Two qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System; and
 - Two diesel generators (DGs) capable of supplying the onsite Class 1E power distribution subsystems

LCO 3.7.8 Two Service Water (SW) trains shall be OPERABLE.

	<u>LCO 3.8.1</u>	<u>LCO 3.7.8</u>
a.	Met	Not Met
b.	Met	Met
c.	Not Met	Met
d.	Not Met	Not Met

ANSWER

c.

QUESTION # 090

Given the following:

- The unit is at 100% Rated Thermal Power.
- Due to required breaker maintenance the electric distribution system is lined up so Bus 2 is supplied by the RAT.

Who is the Shift Manager or Unit Supervisor REQUIRED to NOTIFY because of this change in the electrical distribution system lineup?

- a. American Transmission Company (ATC) and Nuclear Regulatory Commission (NRC).
- b. American Transmission Company (ATC) and Dominion Energy Marketing Inc. (DEMI).
- c. Wisconsin Public Service Corporation (WPSC) and Nuclear Regulatory Commission (NRC).
- d. Wisconsin Public Service Corporation (WPSC) and Dominion Energy Marketing Inc. (DEMI).

ANSWER

b.

QUESTION # 091

Given the following:

- The unit is at 100% Rated Thermal Power
- Control Board IRPI meter for Rod G11 indicates ZERO
- The ROD BOTTOM light for Rod G11 is NOT LIT
- Rod G11 was verified at the proper Control Bank height using incore moveable detectors 1 hour ago
- The IRPI channel for Rod G11 is determined to be INOPERABLE
- The reactor spuriously trips and conditions for Safety Injection are not met
- No other events occur

Which of the following actions will the Unit Supervisor direct?

- a. De-energizing of the Rod Drive MG sets prior to transitioning from E-0, "Reactor Trip or Safety Injection".
- b. Emergency boration will be established using AOP-CVC-001, "Emergency Boration", during performance of the EOPs.
- c. Initiate Safety Injection to establish shutdown boron concentration as directed in the actions of FR-S.1, "Response to Nuclear Generation/ATWS".
- d. Verification of rod G-11 fully inserted into the core using AOP-CRD-001, "Abnormal Rod Drive System", prior to transitioning from the EOPs to the GOPs.

ANSWER

b.

QUESTION # 092

Given the following:

- The unit is operating at 100% Rated Thermal Power.
- Annunciator 47064-Q, Cond Storage Tanks Level High/Low, is LIT due to both tanks being low.
- LCO 3.7.6 Condition A "CSTs inoperable" has been entered due to the low level in the Condensate Storage Tanks (CSTs).

Using the provided reference of 3.7.6 Condensate Storage Tanks (CSTs), which of the following will satisfy Required Action A.1

Verify that...

- a. Service Water Pumps are available and SW supply to AFW pump Valves are OPEN.
- b. Service Water Trains are OPERABLE and able to supply water to the AFW pumps.
- c. Makeup Water Plant started, filling the CSTs as necessary to clear the CST Low level alarm.
- d. Reactor Makeup Storage Tank and one Reactor Makeup Water Pump are available to supply water to the CSTs to provide water for the AFW Pumps.

ANSWER

b.

CSTs
3.7.6

3.7 PLANT SYSTEMS

3.7.6 Condensate Storage Tanks (CSTs)

LCO 3.7.6 The CSTs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CSTs inoperable.	A.1 Verify by administrative means OPERABILITY of backup water supply.	4 hours <u>AND</u> Once per 12 hours thereafter
	<u>AND</u> A.2 Restore CSTs to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 4, without reliance on steam generator for heat removal.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.6.1	Verify usable volume in the CSTs is $\geq 41,500$ gal.	12 hours

QUESTION # 093

Given the following:

- The unit was operating at 100% Rated Thermal Power when the Reactor tripped and Safety Injection initiated due to a Tube Rupture in Steam Generator (SG) 'A'.
- SG 'A' N-16 monitor was indicating Off-Scale HIGH prior to the Reactor trip
- A Loss of Off-Site Power occurred when the Reactor tripped.
- The crew is performing the actions of E-0, "Reactor Trip or Safety Injection".
- Just prior to taking the Main Steam Dump Control Mode Selector Switch to RESET then STM PRESS, the following indications are noted:

<u>Parameter</u>	<u>Value</u>	<u>Trend</u>
Pressurizer Pressure	1140 psig	Stable
Pressurizer Level	0 %	-----
Loop 'A' RCS Cold Leg WR Temp	483°F	Slowly Lowering
Loop 'B' RCS Cold Leg WR Temp	484°F	Slowly Lowering
Containment Pressure	0.3 psig	Stable
SG 'A' Narrow Range Level	2%	Slowly Rising
SG 'B' Narrow Range Level	0 %	-----
SG 'A' Pressure	800 psig	Slowly Lowering
SG 'B' Pressure	800 psig	Slowly Lowering
AFW Flow Header 'A'	120 gpm	Slowly Rising
AFW Flow Header 'B'	110 gpm	Slowly Rising

- After the Main Steam Dump Control Mode Selector Switch is taken to STM PRESS, the Balance of Plant Operator reports that SD5A1, Main Steam Header 'A' Atmos Steam Dump, is OPEN.
- Manual Control of HC-484, Steam Dump Pressure Controller failed to close SD5A1.

What action and procedural flow path will the Unit Supervisor direct? (Directions and Actions are performed in order listed)

1. Close Main Steam Isolation and Bypass Valves, MS-1A, MS-1B, MS-2A, & MS-2B
2. Continue in E-0, "Reactor Trip or Safety Injection"
3. Transition to E-3, "Steam Generator Tube Rupture" to isolate SG 'A'
1. Immediately isolate feed water flow to SG 'A'
2. Continue in E-0, "Reactor Trip or Safety Injection"
3. Transition to E-2, "Faulted Steam Generator Isolation", to isolate steam flow from SG 'A'
1. Continue in E-0, "Reactor Trip or Safety Injection"
2. Transition to E-3, "Steam Generator Tube Rupture", to isolate feed flow to SG 'A'
3. Transition to E-2, "Faulted Steam Generator Isolation", to isolate steam flow from SG 'A'
1. Continue in E-0, "Reactor Trip or Safety Injection"
2. Transition to E-2, "Faulted Steam Generator Isolation", to isolate steam flow from SG 'A'
3. Transition to ECA-3.1, "Steam Generator Tube Rupture With Loss of Reactor Coolant – Subcooled Recovery Desired"

ANSWER

a.

QUESTION # 094

Which of the following is an administrative requirement associated with fuel movement?

- a. A Senior Reactor Operator is required to be in the refueling area to supervise fuel movement.
- b. A Senior Reactor Operator is assigned to oversee the evolution to meet the ICCE category 2 requirements.
- c. A dedicated reactivity Senior Reactor Operator is required to be stationed in the Control Room during fuel movement.
- d. Changes to the refueling plan are required to be reviewed by a Senior Reactor Operator within two months of execution.

ANSWER

a.

QUESTION # 095

In an emergency which of the following is NOT one of the criteria which must be met in order to permit a deviation from license conditions or Technical Specification as allowed by 10 CFR 50.54(x) , "Conditions of Licenses"?

- a. Immediate action is required protect public safety.
- b. Departure is necessary to protect public health and safety.
- c. Performing an equivalent action would result in severe equipment damage.
- d. No action consistent with license condition or Technical Specification which provides equivalent protection is apparent.

ANSWER

c.

QUESTION # 096

Given the following:

- The unit is at 100% Rated Thermal Power.
- Containment Fan Coil Unit (CFCU) 'A' is INOPERABLE for maintenance.
- Internal Containment Spray Pump 'A' is INOPERABLE for maintenance.
- Plant Electricians request Bus 1-61 be removed from service for maintenance.
- Upon SRO review, the Electricians' request is denied

Using the attached references, DETERMINE the reason for the denial?

- a. De-energizing Bus 1-61 will result in a Loss of Safety Function for Containment Cooling.
- b. Both Containment Spray Trains may only be out of service provided that all CFCU's are OPERABLE.
- c. Tech Specs do not provide a required action for de-energizing Bus 1-61 and would require a LCO 3.0.3 entry.
- d. Per PRA configuration control guidelines, only one SSC affecting PRA risk may be removed for maintenance at a time.

ANSWER

a.

Containment Spray and Cooling Systems
3.6.6

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One containment spray train inoperable.	A.1 Restore containment spray train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u>	6 hours
	B.2 Be in MODE 5.	84 hours
C. One containment cooling train inoperable.	C.1 Restore containment cooling train to OPERABLE status.	7 days
D. One fan coil unit in both containment cooling trains inoperable.	D.1 Restore one containment cooling train to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3. <u>AND</u>	6 hours
	E.2 Be in MODE 5.	36 hours

Containment Spray and Cooling Systems
3.6.6

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two containment spray trains inoperable. <u>OR</u> Two containment cooling trains inoperable for reasons other than Condition D.	F.1 Enter LCO 3.0.3.	Immediately

Distribution Systems - Operating
3.8.9

3.8 ELECTRICAL POWER SYSTEMS

3.8.9 Distribution Systems - Operating

LCO 3.8.9 Train A and Train B AC, DC, and AC instrument bus electrical power distribution subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

NOTE

When one or more electrical power distribution subsystems are inoperable solely due to the room cooler being non-functional, entry into associated ACTIONS may be delayed for up to 24 hours provided the associated room temperature is monitored and maintained within the design environmental requirements and the electrical power distribution subsystems in the other train are OPERABLE.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more AC electrical power distribution subsystems inoperable.	<p>NOTE</p> <p>Enter applicable Conditions and Required Actions of LCO 3.8.4, "DC Sources - Operating," for DC sources made inoperable by inoperable power distribution subsystems.</p> <p>A.1 Restore AC electrical power distribution subsystem(s) to OPERABLE status.</p>	8 hours
B. One or more AC instrument buses inoperable.	B.1 Restore AC instrument bus(es) to OPERABLE status.	2 hours
C. One DC electrical power distribution subsystem inoperable.	C.1 Restore DC electrical power distribution subsystem to OPERABLE status.	2 hours

Distribution Systems - Operating
3.8.9

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Be in MODE 3.	6 hours
	<u>AND</u> D.2 Be in MODE 5.	36 hours
E. Two or more electrical power distribution subsystems inoperable that result in a loss of safety function.	E.1 Enter LCO 3.0.3.	Immediately

QUESTION # 097

During the use of NOP-SER-001, "Control Room Sequential Event Recorder" to disable an SER point, when is a 10CFR50.59 Review REQUIRED to be performed?

When . . .

- a. the disabled SER point is described in the KPS USAR.
- b. directed to disable the SER point during the performance of a maintenance procedure.
- c. disabling the SER point because the affected annunciator/sensor circuit is malfunctioning.
- d. disabling the SER point is a reasonable step in repairing or troubleshooting the affected circuit.

ANSWER

a.

QUESTION # 098

Given the following:

- The unit is in MODE 4 with a startup in progress following a refueling outage.
- Steam Generator Blowdown is in Mode I, and was initiated 18 hours ago.
- Chemistry Reports that confirmed and validated samples for 'A' & 'B' Steam Generator Total Gamma Activities are 8×10^{-6} $\mu\text{Ci/ml}$ and 2.8×10^{-5} $\mu\text{Ci/ml}$ respectively.
- Previous Steam Generator samples for both were $< 1 \times 10^{-6}$ $\mu\text{Ci/ml}$ Total Gamma Activity.
- R-19, Steam Generator Blowdown Monitor, has been constant at 7.02E1 cpm for the past 4 days.

What actions should the Unit Supervisor direct in response to the rise in activity of the Steam Generators?

- a. Secure Blowdown to stop the release to Auxiliary Building standpipe

AND

Enter AOP-RC-004, "Steam Generator Tube Leak", when Chemistry reports Primary to Secondary leak rate is > 5 gpd

- b. Switch Blowdown to Startup Mode II to align blowdown liquid to the Steam Generator Blowdown Tanks

AND

Enter AOP-RHR-002, "Shutdown Loss of Coolant Accident"

- c. Maintain Blowdown Mode I and direct Steam Generator Sampling for R-19 out of service per the requirements of the Offsite Dose Calculation Manual

AND

Enter AOP-RC-004B, "POST SG Tube Leak Cooldown Using Blowdown", when Chemistry reports Primary to Secondary leak rate is > 1 gpd

- d. Switch Blowdown to Mode III to remove the activity from the blowdown liquid prior to discharging it to the Auxiliary Building standpipe

AND

Enter E-1, "Loss of Reactor or Secondary Coolant"

ANSWER

a.

QUESTION # 099

Given the following:

- The unit is in MODE 6
- Personnel are on station in preparation for fuel movement
- Refueling cavity level is 23 ft above the reactor vessel flange
- Decay heat removal is provided by RHR
- RCS Pressure is at atmospheric
- RHR suction temperature is 110°F
- 'A' RHR Pump is running
- RHR/CVC spectacle flange is OPEN
- RHR-210, RHR/CVC Inlet Isolation, is OPEN
- RHR-211, RHR/CVC Outlet Isolation, is OPEN

3 minutes later the following is noted:

- 47031-Q, Containment Sump A Level Hi alarms followed by 47031-P, Containment Sump A Level Hi-Hi alarms
- The Refueling SRO reports to the Control Room, the refueling cavity level is visibly lowering

Which procedure will the Unit Supervisor enter to perform actions which to MITIGATE this condition?

- a. E-1, "Loss of Reactor or Secondary Coolant".
- b. AOP-RC-002, "Abnormal Refueling Water Level".
- c. AOP-FH-001, "Loss of Reactor Cavity Inventory During Fuel Movement".
- d. AOP-MDS-001, "Abnormal Operation of Miscellaneous Drains and Sumps".

ANSWER

c.

QUESTION # 100

Given the following:

- The unit is in MODE 3.
- 1053 Fire Works Smoke Detector alarm is received for Safeguards Alley area.
- 1054 The Shift Manager receives a report from the Nuclear Security Shift Supervisor that at 1052 there was a security condition that does not involve a HOSTILE ACTION occurred when an unauthorized person was apprehended exiting Safeguard Alley.
- 1056 AFW Pump 'A' trips.
- 1057 The NAO reports a fire in the AFW Pump 'A' area and visible damage to the AFW Pump 'A'.
- 1107 Report from the Fire Brigade Leader of "The fire is out" is communicated to the Control Room.

Using the provided reference, DETERMINE which of the following classifications meet the requirements of the Emergency Plan.

Classification of ____.

- a. Site Area Emergency per HS4.1
- b. Unusual Event per HU2.1
- c. Alert per HA4.1
- d. Alert per HA2.1

ANSWER

d.

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
Hazards	Fire or Explosion	None	None	<div>FIRE or EXPLOSION Affecting the Operability of Plant Safety Systems Required to Establish or Maintain Safe Shutdown</div> <div><input type="checkbox"/> HA2.1 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>FIRE or EXPLOSION in any of the following areas (Table H-1) AND Affected safety system parameter indications show degraded performance or plant personnel report VISIBLE DAMAGE to permanent structures or equipment needed for safe shutdown</div>	<div>FIRE Within PROTECTED AREA Boundary Not Extinguished Within 15 Minutes of Detection</div> <div><input type="checkbox"/> HU2.1 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>FIRE in buildings or areas contiguous to any Table H-1 Safe Shutdown/VITAL Area not extinguished within 15 minutes of control room notification or verification of a control room alarm</div>
	Toxic and Flammable Gas	None	None	<div>Release of Toxic or Flammable Gases Within or Contiguous to a VITAL AREA Which Jeopardizes Operation of Systems Required to Maintain Safe Operations or Establish or Maintain Safe Shutdown</div> <div><input type="checkbox"/> HA3.1 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>Report or detection of toxic gases within or contiguous to a Safe Shutdown/VITAL AREA (Table H-1) in concentrations that may result in an atmosphere IMMEDIATELY DANGEROUS TO LIFE AND HEALTH (IDLH)</div> <div><input type="checkbox"/> HA3.2 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>Report or detection of gases in concentration greater than the LOWER FLAMMABILITY LIMIT within or contiguous to a Safe Shutdown/VITAL AREA (Table H-1)</div>	<div>Release of Toxic or Flammable Gases Deemed Detrimental to Normal Operation of the Plant</div> <div><input type="checkbox"/> HU3.1 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>Report or detection of toxic or flammable gases that has or could enter the site area boundary in amounts that can affect NORMAL PLANT OPERATIONS</div> <div><input type="checkbox"/> HU3.2 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>Report by Local, County or State Officials for evacuation or sheltering of site personnel based on an offsite event</div>
	Security	<div>HOSTILE ACTION resulting in loss of physical control of the facility</div> <div><input type="checkbox"/> HG1.1 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>A HOSTILE ACTION has occurred such that plant personnel are unable to operate equipment required to maintain safety functions</div> <div><input type="checkbox"/> HG1.2 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>A HOSTILE ACTION has caused failure of Spent Fuel Cooling Systems and IMMINENT fuel damage is likely for a freshly off-loaded reactor core in the Spent Fuel Pool</div>	<div>HOSTILE ACTION within the PROTECED AREA</div> <div><input type="checkbox"/> HS4.1 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>A HOSTILE ACTION is occurring or has occured within the PROTECTED AREA as reported by Security Supervision</div>	<div>HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat</div> <div><input type="checkbox"/> HA4.1 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by Security Supervision</div> <div><input type="checkbox"/> HA4.2 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>A validated notification from NRC of an airliner attack threat within 30 minutes of the site</div>	<div>Confirmed SECURITY CONDITION or threat which indicates a potential degradation in the level of safety of the plant</div> <div><input type="checkbox"/> HU4.1 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>A SECURITY CONDITION that does NOT involve a HOSTILE ACTION as reported by the Security Supervision</div> <div><input type="checkbox"/> HU4.2 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>A credible site specific security threat notification</div> <div><input type="checkbox"/> HU4.3 <input type="checkbox"/> OP <input type="checkbox"/> HSB <input type="checkbox"/> HSD <input type="checkbox"/> ISD <input type="checkbox"/> CSD <input type="checkbox"/> REF <input type="checkbox"/> DEF</div> <div>A validated notification from NRC providing information of an aircraft threat</div>

Table H-1 Safe Shutdown/VITAL Areas
<div>- Shield Building (Reactor Building)</div> <div>- Auxiliary Building</div> <div>- Safeguards Alley</div> <div>- Diesel Generator Rooms (includes "A" Diesel Room to Screen House Tunnel)</div> <div>- Screenhouse/Forebay</div> <div>- Technical Support Center Basement</div> <div>- Control Room</div> <div>- Control Room AC Equipment Room</div> <div>- Relay Room</div> <div>- Safeguards Battery Rooms</div>

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